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

OFFICE OF THE CHAIRMAN

December 9, 1985

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

Dear Mr. President:

Enclosed is the NRC report on abnormal occurrences at licensed nuclear facilities, as required by Section 208 of the Energy Reorganization Act of 1974 (PL 93-438), for the second calendar quarter of 1985.

In the context of the Act, an abnormal occurrence is an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The report states that for this report period, there were three abnormal occurrences at the nuclear power plants licensed to operate. These events involved, respectively, (1) inoperable safety injection pumps, (2) significant deficiencies in reactor operator training and material false statements, and (3) loss of main and auxiliary feedwater systems. There were four abnormal occurrences at the other NRC licensees. Three events involved diagnostic or therapeutic medical misadministrations; the other involved a breakdown in management controls. There was one abnormal occurrence reported by an Agreement State; the event involved overexposures of a radiographer and an assistant radiographer.

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

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

Sincerely,

Nunzio J. Palladino

Nunzio J. Palladino
Chairman

Enclosure:
Report to Congress on
Abnormal Occurrences
NUREG-0090, Vol. 8, No. 2

NUREG-0090
Vol. 8, No. 2

Report to Congress on Abnormal Occurrences

April - June 1985

U.S. Nuclear Regulatory Commission

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ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from April 1 to June 30, 1985.

The report states that for this reporting period, there were three abnormal occurrences at the nuclear power plants licensed to operate. These events involved, respectively, (1) inoperable safety injection pumps, (2) significant deficiencies in reactor operator training and material false statements, and (3) loss of main and auxiliary feedwater systems. There were four abnormal occurrences at the other NRC licensees. Three events involved diagnostic or therapeutic medical misadministrations; the other involved a breakdown in management controls. There was one abnormal occurrence reported by an Agreement State; the event involved overexposures of a radiographer and an assistant radiographer.

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

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PREFACE

INTRODUCTION

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

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

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement states, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period from April 1 to June 30, 1985.

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 protection. These multiple levels can

be achieved and maintained through regulations which specify requirements which will assure the safe use of nuclear materials. The regulations include design and quality assurance criteria appropriate for the various activities licensed by NRC. An inspection and enforcement program helps assure compliance with the regulations.

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

REPORTABLE OCCURRENCES

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

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

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

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

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

AGREEMENT STATES

Section 274 of the Atomic Energy Act, as amended, authorizes the Commission to enter into agreements with States whereby the Commission relinquishes and the States assume regulatory authority over byproduct, source and special nuclear

materials (in quantities not capable of sustaining a chain reaction). Comparable and compatible programs are the basis for agreements.

Presently, information on reportable occurrences in Agreement State licensed activities is publicly available at the State level. Certain information is also provided to the NRC under exchange of information provisions in the agreements.

In early 1977, the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly 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
APRIL - JUNE 1985

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the second calendar quarter of 1985. As of the date of this report, the NRC had determined that the following events were abnormal occurrences.

85-5 Inoperable Safety Injection Pumps

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

Date and Place - On December 28, 1984, Consolidated Edison Company of New York (the licensee) declared all three safety injection pumps inoperable at Indian Point Unit 2. The plant, which utilizes a Westinghouse-designed pressurized water reactor, is located in Westchester County, New York.

Nature and Probable Consequences - On December 28, 1984 with the reactor critical, operators attempted to top off the emergency core cooling system (ECCS) accumulator tanks by running safety injection (SI) pump #23. The pump developed a discharge head of 1100 psig, but then dropped to about 700 psig (which corresponds to the pressure in the accumulator tanks). The pump was secured and SI pump #22 was started to top off the accumulator tanks.

After topping off two of the four accumulator tanks, and while in the process of topping off the third, the #22 pump was observed to drop in discharge pressure from 1500 psig to the accumulator tank pressure of about 700 psig. This pump was also secured.

The piping to the third SI pump #21 precluded its use to top off the accumulator; however, an attempt was made to manually turn over the pump. This attempt was unsuccessful. Therefore, the licensee declared all three SI pumps inoperable. Because the reactor was critical at this time, the licensee manually scrammed the control rods to shutdown the reactor and implemented an orderly approach to cold shutdown conditions. At no time during the event were the SI pumps required to perform their safety function. However, with all three SI pumps inoperable for a period of up to nine days (one day with the reactor critical), the automatic capability to deal with a design basis accident (i.e., steam line break) was significantly degraded.

As discussed further below, investigation showed that SI pumps #22 and #23 were degraded due to partial blockage of their suction path by solidified boric acid, together with gas entrapment causing the pumps to bind. For SI pump #21, total blockage of its suction line occurred due to solidified boric acid.

The licensee has commissioned Westinghouse to perform analysis to determine the feasibility of either removing the BIT or reducing the boric acid concentration of the BIT.

Background Information

The three "intermediate" head SI pumps are part of the plant's ECCS. The pumps have a nominal 1700 psig discharge pressure. There is no installed ECCS capability at full system pressure.

Boron injection tanks (BITs) were installed in Westinghouse plants as a measure to ensure that sufficient negative reactivity would be inserted during a steam line break event to compensate for the positive reactivity addition resulting from the rapid cooldown. This ensures complete reactor shutdown and thus, minimizes the potential for fuel failures. The BITs contain a high concentration (20,000 ppm boron) of boric acid. Upon receipt of a safety features actuation, the SI pumps sweep the contents of the BIT into the reactor coolant system. Before the BIT is emptied, the suction of the pumps switches automatically to the refueling water storage tank (RWST) which contains boric acid with a concentration of 2,000 ppm boron.

The Indian Point Unit 2 plant is unusual in that the BIT discharge line is aligned to the suction of the SI pumps. Nitrogen overpressure in the BIT provides the motive force to inject its contents into the SI pumps suction header in advance of the RWST contents when the isolation valves are signaled to open. In the usual Westinghouse-designed plant however, the BIT is located on the discharge side of high head charging pumps; in this case, during the initial stages of safety injection, the charging pumps take suction from the RWST and discharges through the BIT, sweeping the latter's contents into the reactor coolant system. In this case, the highly concentrated boric acid from the BIT does not flow through pumps as it does in Indian Point Unit 2.

In highly concentrated solutions, boric acid will precipitate and solidify if the solution is not heated. The BIT in Indian Point Unit 2 utilizes electric tank heaters and line heat tracing to keep the temperature of the boric acid above the solubility limit; however, the SI pumps are not heat traced.

<u>Cause or Causes</u> - The cause is attributed to boric acid precipitating from a highly concentrated solution, solidifying and preventing suction flow. In addition, a gas caused the SI pumps to bind. There was total blockage of pump #21 by solidified boric acid and partial blockage of the #22 and #23 pumps. While venting the pumps, only gas was emitted from pump #21, only water flowed from pump #23, and both gas and water were vented from pump #22.

The source of the boric acid is believed to be the BIT. The BIT was leaking past closure valves 1822A and 1822B which isolate the BIT from the SI pump suction. The BIT solution apparently precipitated and solidified because the SI pumps are not heat traced.

Incomplete flushing of the SI pumps following SI actuation could also result in the BIT contents reaching the SI pumps. On December 19, 1984, the SI pumps operated during a plant trip. The contents of the BIT discharged into the SI lines, but did not inject into the reactor coolant system because the reactor coolant system did not depressurize below the discharge head of the SI pumps.

Following the trip, the SI pumps were flushed in accordance with plant Procedure E-4, "Recovery From a Spurious Safety Injection"; however, the procedure did not refer to SOP 10.11, "Filling, Draining, Flushing SI System," a procedure that more clearly defines the SI pump flushing and BIT filling methods.

While venting SI pump #21, the licensee took a gas sample for analysis and found that the major constituent was nitrogen (97%). There are several potential sources for nitrogen gas, i.e., (1) the isolation valve seal water system (IVSWS), which injects nitrogen between some SI valves in order to seal them to provide improved containment isolation following an accident, (2) the nitrogen cover gas in the ECCS accumulators, and (3) the nitrogen cover gas in the BIT.

The licensee initially believed that the most likely source of the gas in the pumps was from the nitrogen cover in the BIT. The licensee commissioned their consultant at Lehigh University (which has a scale model of the licensee's SI system) to perform theoretical and experimental analyses to determine the feasibility of the BIT being the source of the gas in the pumps. The consultant concluded that the BIT was not likely to be the source of the gas; therefore, the licensee is continuing its investigations.

Actions Taken to Prevent Recurrence

Licensee - The BIT discharge line boric acid concentration is being monitored on a daily basis and is flushed upon detection of increasing boric acid. The SI pumps are vented daily and monitored for gas. The emergency procedure for recovery from a spurious safety injection has been clarified to provide for adequate flushing of the BIT discharge line. The IVSWS nitrogen header to the SI system has been isolated. The licensee is also preparing a Technical Specification Amendment request to allow removal of the BIT (see Editor's Note below).

NRC - The NRC monitored the licensee's response to this event and confirmed completion of the corrective and preventive actions taken as described above. The NRC performed an inspection of the circumstances associated with the event. The results are contained in NRC Inspection Report 50-247/84-33 dated February 14, 1985 (Ref. 1).

Editors Note

Over the past several years, the analysis methods for calculating the consequences of steam line break have improved. These revised calculations demonstrate that the negative reactivity that must be added to meet current requirements is lower than originally thought. Consequently, the need for highly concentrated boron injection may be reduced or eliminated.

Because of various operational problems and possible safety risks associated with the high boron concentrations in the BITs, and the improved calculations, many licensees with Westinghouse plants have requested that they be allowed to either physically remove the BIT from the safety injection piping, or at least reduce boron concentrations in the tank to the levels safely used in other sections of the safety injection piping and RWST (e.g., to 2,000 ppm). To support their requests, licensees have submitted new analyses of the steam line break event that demonstrated the regulatory criteria (i.e., 10 CFR §100 guidelines dose values) were met.

The NRC staff has reviewed these analyses and has approved these requests. The plants which have received approval for removal of the BIT, or changes to Technical Specification requirements on boron concentration in the BIT, include: Turkey Point Units 3 and 4, Surry Units 1 and 2, Beaver Valley Unit 1, McGuire Unit 1 (Unit 2 was licensed without a BIT), Catawba Units 1 and 2, Callaway Unit 1, Farley Units 1 and 2, Trojan, South Texas and Harris (the latter two plants are presently under licensing review).

In addition to the above licensing action, the NRC Office of Nuclear Reactor Regulation is currently working on two initiatives which address the BIT issue generically. A generic letter to the appropriate licensees is being prepared which informs them of staff approval of the improved steam line break analysis methods and encourages them to reevaluate the need for maintaining high concentrations of boron in the BIT. In addition to this, a study has been initiated to determine whether or not a stronger position on the BIT removal issue is appropriate; i.e. whether higher boron concentrations in the BIT should be prohibited rather than simply discouraged. This study will also be used to identify the schedule and the appropriate level of resources to be allocated for resolution of this issue.

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

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85-6 <u>Significant Deficiencies in Reactor Operator Training and Material False Statements</u>

The following information pertaining to this event is also being reported concurrently in the <u>Federal Register</u>. Appendix A (see Example 11 of "For All Licensees") of this report notes that serious deficiencies in management or procedural controls in major areas can be considered an abnormal occurrence. In addition, general criterion 3 notes that major deficiencies in use of, or management controls for licensed facilities can be considered an abnormal occurrence.

<u>Date and Place</u> - By letter of June 3, 1985 (Ref. 2), the NRC issued to Mississippi Power and Light Company (MP&L), licensee of the Grand Gulf facility, a Notice of Violation and Proposed Imposition of Civil Penalties for identified deficiencies in the reactor operator training program and for making material false statements to the NRC. Applications for reactor operator licenses containing apparently false information were submitted to the NRC in September 1981, March 1982, and May 1982. The Grand Gulf plant utilizes a General Electric-designed boiling water reactor and is located in Claiborne County, Mississippi.

<u>Nature and Probable Consequences</u> - The June 3, 1985 NRC letter identified serious failures to comply with NRC regulatory requirements at Grand Gulf. Most of the violations pertained to the reactor operator (RO) and senior reactor operator (SRO) training program, including (1) inadequate procedures, instructions, and procedural controls; (2) training certifications which contained material false statements; and (3) failure to correct false submittals once the licensee became aware of them. The violations, classified as high as Severity Level I, were documented by special inspections by the Region II Office and by investigations by the NRC Office of Investigations.

Discrepancies in documentation of operator training were identified during a special training assessment conducted in February 1983 and a special safety inspection conducted by the Region II Office during August and September 1983. The Region II staff evaluated these inspections and concluded that these discrepancies were not limited to documentation errors. At Region II's request, the Office of Investigations conducted investigations during the period of October 18, 1983 through May 9, 1984. The investigation included a review of the circumstances surrounding the submittal of false and undocumented information on operator license applications. As a result of these inspections and the investigation efforts, significant failures to comply with NRC regulatory requirements were identified.

The inspection and investigation findings demonstrate that the program for training ROs and SROs at the Grand Gulf facility had not been established in accordance with commitments made in the Final Safety Analysis Report (FSAR) and as required by NRC regulations. The investigation also determined that 46 applications for SRO and RO licenses, containing certification by MP&L that each individual applicant had completed required training or courses of instruction, contained material false statements. The information provided was false in that the amount of training actually completed was less than that described in the operator license applications. The information was material because had the complete and accurate information been known to the NRC, the applicants would not have been permitted to participate in the NRC licensing examination and, consequently, would not have received licenses. In addition, even after MP&L officials became aware in 1982 that false information had been submitted, they failed to notify the NRC or to correct the submittals. This constitutes a separate material false statement by omission.

Item 1 in the Notice of Violation and Proposed Imposition of Civil Penalties addresses the training program inadequacies. In this case, MP&L had not established an effective program for assuring commitments made in the FSAR were implemented in the operator license training program. Specifically, MP&L delegated control of the training program to a contractor and did not exercise adequate oversight of training activities. This contributed directly to the failure to meet the commitment for comprehensive and adequate training of operator license candidates.

Items 2 and 3 of the Notice concern the material false statements. The NRC requires extraordinary care be taken to assure information provided in applications is complete and accurate. MP&L did not adequately verify the information prior to its submittal to the NRC, vigorously implement a program to identify and document the false information after being informed of its existence by a licensee employee, or inform the NRC that false information had been submitted once it became aware that the submittals contained false information.

Item 4 of the Notice addresses a procedural violation involving failure of a mechanical maintenance supervisor to correctly complete a practical factors book for a mechanic. The cause of this violation was that inadequate instructions on how to accomplish the tasks were provided to supervisors responsible for following the procedures.

Item 3 of the Notice was characterized as a Severity Level I violation because it was a knowing failure to correct previously submitted false information. Items 1 and 2 of the Notice were classified as Severity Level II, and Item 4 was classified as Severity Level IV.

Had these problems gone undetected, the probable consequences are that the plant would have continued operation with operators not fully trained to accomplish their jobs. These were serious violations and positive corrective actions were not taken until the NRC became involved. The violations occurred in careless disregard for NRC requirements.

Cause or Causes - The cause of these occurrences was failure to exercise management control. The licensee management gave low priority to the reactor operator training program, and relied heavily on unmonitored contractors to train and certify completion of training.

Actions Taken to Prevent Recurrence

Licensee - As a result of the 1983 management meetings (described below) held at the Region II Office, MP&L committed to conduct a review of the previous training of all licensed operators, shift technical advisors, and on-shift operations advisors. Certain operators were removed from licensed duties until they could be retrained and retested. These commitments were confirmed by a letter dated December 5, 1983.

As a result of these commitments, MP&L examined each operator on each of 68 systems listed on the Grand Gulf licensed operator qualification card. These examinations were monitored by MP&L management, representatives of two other utilities, the Nuclear Steam Supply System vendor (General Electric), and the NRC. At the completion of this examination process, the records of the operators were reviewed by a Grand Gulf recertification board consisting of plant management. The board examined operator training records, the results of the examinations, and conducted additional oral examinations as necessary. Out of 27 individuals examined by the board, one was found to be unqualified and three needed training.

Region II conducted licensed operator recertification and walk through examinations in February 1984 after each licensed operator had undergone the MP&L examinations. The results of the independent NRC recertification examination were that 23 of the 26 operators passed. The three who failed have been removed from licensed duties. Management was responsive and aggressively sought improvements to plant training programs. A change of management personnel in the plant training staff improved the overall training program administration. This reorganization resulted in better documentation of training, clearer standards of acceptable performance for both students and training staff, and improved adherence to regulations, procedures and commitments.

The consolidation of the entire training organization into the new training facility has also improved the quality of training. These actions provide reasonable assurance that operators presently at the controls of the facility have met NRC requirements for training.

NRC - As previously mentioned, the discrepancies in documentation of operator training were identified during a special training assessment conducted in February 1983 and a special safety inspection conducted by the NRC Region II Office during August and September 1983. Investigations regarding submittal of false and undocumented information on operator license applications were made by the NRC Office of Investigations during the period of October 18, 1983 through May 9, 1984.

Management meetings to discuss the NRC Region II Office findings were held in the Region II Office on September 23, October 12, November 11 and 18, 1983. On June 3, 1985, the NRC issued the previously mentioned Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $500,000 (Ref. 2). The licensee responded on September 12, 1985. The NRC is reviewing the licensee's response to assure that all of the issues are satisfactorily resolved.

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

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85-7 Loss of Main and Auxiliary Feedwater Systems

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the second general criterion) of this report notes that a major degradation of essential safety-related equipment can be considered an abnormal occurrence. In addition, Example 11 of "For All Licensees" notes that serious deficiencies in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - On June 9, 1985, the Davis-Besse Nuclear Power Plant experienced a complete loss of main and auxiliary feedwater for about 12 minutes during an event involving an automatic shutdown from operation at 90% power. Davis-Besse utilizes a Babcock & Wilcox - designed pressurized water reactor. The plant is operated by Toledo Edison Company (the licensee) and is located in Ottawa County, Ohio.

The event involved several equipment malfunctions and extensive operator actions, including operator actions outside the control room.

Nature and Probable Consequences - Early in the morning on June 9, 1985, the plant was operating at 90% power with the No. 1 main feedwater pump (MFP) operating in automatic and the No. 2 MFP in manual control. This configuration was established to limit the susceptibility of the No. 2 MFP to automatic control problems which had affected the operation of both MFPs since April 1985. (At the time of the event, troubleshooting had neither identified the root cause of, nor resolved the root cause of the problems.) The plant's integrated control system (ICS) and associated instrumentation were automatically monitoring and controlling the thermo-hydraulic balance between the reactor coolant system and the secondary coolant system.

At 1:35 a.m., a control system failure caused the No. 1 MFP to trip on overspeed. Since the No. 2 MFP was in manual control, it did not respond to the ICS demand to automatically increase feedwater flow. The control room operators thus manually increased the No. 2 MFP speed (to increase feedwater flow). Meanwhile, the ICS attempted automatically to run back reactor/turbine power by inserting control rods. However, these actions were insufficient to prevent the reactor coolant pressure from rising excessively. At about 80% reactor power, the pressure reached the high pressure reactor trip set point of 2,300 psig (normal operating pressure is 2,150 psig). The reactor trip (which occurred 30 seconds after the trip of the No. 1 MFP) automatically tripped the turbine. As part of the normal reactor trip procedure, the operator isolated the reactor coolant system (RCS) letdown and started a second RCS makeup pump in anticipation of the pressurizer inventory decreasing as a result of the reactor trip.

The MFPs at Davis-Besse are turbine driven by steam taken from downstream of the main stream isolation valves (MSIVs). As long as the MSIVs remain open and the steam generators are producing sufficient steam to drive the MFPs to generate a discharge pressure greater than steam generator pressure, the MFPs can provide feedwater to the steam generators.

However, one second after reactor/turbine trip, the steam and feedwater rupture control system (SFRCS) activated due to a spurious low steam generator water level signal. The signal was spurious since the actual steam generator level was not low at the time. A partial actuation of the SFRCS occurred which closed both MSIVs (both were closed within six seconds after the SFRCS actuation), isolating the main steam supply from the MFPs. Three seconds after the partial actuation, the SFRCS automatically reset. However, the MSIVs remained closed. At this time, the water level in the steam generators was a normal post-trip level of 35 inches. About four minutes later, the No. 2 MFP discharge pressure dropped below the steam generator pressure which terminated main feedwater flow.

After the MSIVs closed, steam pressure in the steam generators increased until the main steam valves lifted to relieve the pressure. Without feedwater, as in this event, the once-through steam generators (OTSGs) in Babcock & Wilcox-designed plants can boil dry in as little as three minutes; therefore, it is vital to restore some feedwater quickly to avoid loss of this heat transfer path from the primary system to the secondary system. Even though the reactor was shut down, heat continues to be generated by the radioactive decay of the fission products in the fuel. The SFRCS will automatically actuate the auxiliary feedwater system (AFWS) when the steam generator level decreases to a low water level (26.5 inches) to maintain the heat transfer to the OTSGs.

The AFWS utilizes two turbine driven pumps with their motive force provided by steam taken from the main steam system at a point upstream of the MSIVs. Therefore, even with the MSIVs closed, there is a motive force for these pumps as long as steam is being generated in the OTSGs. About 5½ minutes after reactor trip, the water level in OTSG No. 1 reached the low level setpoint of 26.5 in. which actuated channel No. 1 of the SFRCS. This started AFW pump No. 1 and initiated alignment of it to OTSG No. 1. Before this automatic actuation, however, an operator who realized an automatic actuation was imminent (due to the decreasing water levels in the OTSGs) went to the back panel to manually trip the SFRCS. The operator had been trained to manually trip certain systems which he felt were going to trip automatically. For example, tripping the SFRCS early could conserve steam generator water inventory.

Thus, four seconds after the automatic initiation of SFRCS started aligning AFW pump No. 1, the operator manually tripped the SFRCS. However, he inadvertently pushed the wrong two buttons (i.e. actuating the SFRCS on low OTSG pressure rather than on low OTSG water level). This signalled the SFRCS that both OTSGs had experienced a steamline break and the system responded, as designed, to close the isolation valves between each OTSG and its associated AFW train. Therefore, even though the AFWS was started and aligned to provide emergency feedwater, both OTSGs were isolated from all auxiliary feedwater. Within less than a minute after the operator pushed the wrong buttons, both AFW pumps tripped on overspeed (a common-mode failure of the AFWS).

The control room operator quickly determined that the valves in the AFWS were improperly aligned. He reset the SFRCS actuation on low pressure and retripped it on low level. This action commanded the SFRCS to realign itself such that

each AFW pump would deliver flow to its associated OTSG. This action should also have automatically opened the AFWS isolation valves. However, these valves did not open (common mode failure of the AFWS). Backup manual attempts to open them from the control room were also unsuccessful.

Operators were sent to locally start the AFW pumps and to open the AFWS isolation valves. The assistant shift supervisor made the decision to place the startup feedwater pump (SUFP) in service to supply feedwater to the OTSGs.

The SUFP is part of the main feedwater system, but is isolated from the system during power operations. The pump is motor driven and therefore is not dependent upon steam for its motive force. The SUFP is normally used during plant startup. When the plant is in Mode 3 (hot standby), the SUFP maintains OTSG water level. When reactor power reaches about 1%, a main feedwater pump is placed in service and the SUFP is shut down. Per a commitment to the NRC, the SUFP is then isolated by locally closing four manual valves and removing the fuses from the motor control circuit. The non-safety-related SUFP does not meet high energy line break design requirements and under postulated conditions its failure could affect the performance of the safety-related AFWS.

While the assistant shift supervisor was making the SUFP operational, operators were opening the AFWS isolation valves to make the AFW system operational and were working to open the pumps' trip throttle valves so that steam could enter the pumps' turbines. They experienced some difficulties with the latter efforts.

Meanwhile, the OTSGs continued to boil dry. Pressure and temperature in the reactor coolant system continued to rise due to insufficient heat transfer to the OTSGs. About 13 minutes after reactor trip, RCS pressure reached the setpoint (2,425 psig) of the pressurizer pilot operated relief valve (PORV). During about the next two minutes, the PORV opened three times, relieving pressurizer pressure to the quench tank. After the third opening, the PORV failed to close. A rapid RCS depressurization occurred. The reactor operator noticed the depressurization but did not attribute it to the PORV. Nevertheless, he closed the PORV block valve and took other actions as precautionary measures. Two minutes later, he opened the block valve to ensure the PORV was available. Fortunately, the PORV valve had closed by itself while the block valve was closed.

Plant emergency procedures stipulate that when both OTSGs are "dry", make-up/high pressure injection (MU/HPI) cooling (known as "feed-and-bleed") must be initiated for decay heat removal. An OTSG is considered "dry" when its pressure falls below 960 psig and is decreasing, or when its water level is below eight inches on the startup range instrumentation. Due to inadequate and inoperable instrumentation, the operators may not have recognized that at about 1:47 a.m., both OTSGs were dry. The shift supervisor was aware of the reactor core status and that MU/HPI cooling may be necessary. Even though other personnel recommended that MU/HPI be initiated [based on the elevated primary coolant temperature of 591°F (normal post-trip temperature is about 550°F), and delays in initiating feedwater flow], the shift supervisor decided to wait and see if feedwater became available within a short time. He was concerned that MU/HPI cooling would complicate matters further by creating an adverse environment in the containment.

At about 16 minutes after the reactor trip, the SUFP started to provide feedwater, nearly all of it directed to the No. 1 OTSG. The No. 1 OTSG pressure increased sharply and its water level slowly increased. At about two minutes

and four minutes later, AFW train Nos. 2 and 1 respectively became available and produced significant feedwater flow to their respective OTSGs. However, the flow from the AFW trains (which was being controlled locally by operators manipulating the trip throttle valves of the AFW pumps) was excessive and the RCS experienced an overcooling transient. RCS temperature peaked at about 592°F (at about 1:53 a.m.) and decreased sharply to 540°F in about six minutes. Overfilling of the OTSGs decreased RCS pressure towards the setpoint (1650 psig) of the safety features actuation system. To avoid this actuation, the operators aligned a portion of the emergency core cooling system to the primary system to inject borated water to raise RCS pressure. However, by the time injection took place, RCS pressure increased so only about 50 gallons of borated water was actually injected.

At about 1:58 a.m., the No. 1 AFW pump suction transferred spuriously from the condensate storage tank (CST) to the service water system. The operator manually realigned the suction back to the CST. This malfunction was not significant in regard to event recovery, but the problem had occurred previous to this event and had not been corrected. Other malfunctions which occurred during the event (and had occurred previously and had not been properly repaired) included loss of a source range nuclear instrument after reactor trip (the other channel was already inoperable before the trip), and tripping of control room ventilation system into its emergency recirculation mode. Both channels of the safety parameter display system were inoperable prior to, and throughout the event.

By 2:04 a.m., plant conditions were essentially stable which terminated the event. This was about 30 minutes, and twelve equipment malfunctions (including multiple common mode failures), after the event began.

A total loss of feedwater is considered a significant event. Unless prompt and effective recovery actions are taken, severe consequences could occur (e.g., fuel damage, break of primary system, significant release of radioactivity). For this event, compensatory actions were complicated due to many equipment malfunctions and various operator errors. Nevertheless, the operators were successful in bringing the plant to a stable shutdown and in preventing any abnormal releases of radioactivity and any major damage to the plant.

Cause or Causes - As described in more detail above, the event was initiated, and recovery actions made complex, by multiple equipment malfunctions (including several common-mode failures) and a number of personnel errors. The root causes of the various malfunctions and other problems associated with the event are under extensive study.

Actions Taken to Prevent Recurrence

Licensee - The licensee has undertaken an extensive study (including testing programs) of the multiple failures associated with the event to determine root causes and to take effective corrective actions to minimize recurrence. While some tentative conclusions have been made, they may be modified as the ongoing studies continue.

The licensee is keeping the NRC aware of the results of their studies and testing programs. Results of the licensee's actions taken as of July 9, 1985, are contained in their Licensee Event Report (LER) No. 85-13 (Ref. 3). In addition

to determining the root causes of the equipment malfunctions and the appropriate corrective actions required, other issues (e.g., operational, management, procedural deficiencies; equipment design, testing, maintenance deficiencies; etc.) may require extensive licensee efforts.

In response to an August 14, 1985 NRC letter to the licensee (Ref. 4) discussed below, on September 10, 1985 the licensee submitted a course of action (Ref. 5) to address numerous areas of concern identified by the NRC.

NRC - Upon being notified of the event, the NRC Resident Inspectors for the plant arrived shortly thereafter. They observed licensee actions to assure the plant remained in a stable condition and began an initial investigation of the circumstances associated with the event. Personnel from the NRC Region III Office also went to the site after the event. On June 11, 1985, the responsibility for the incident investigation was assumed by a special NRC Incident Investigation Team described below.

On June 10, 1985, the Regional Administrator of the NRC Region III Office forwarded a Confirmatory Action Letter to the licensee (Ref. 6) indicating, among other things, that the licensee would not perform any additional work on equipment that malfunctioned during the event until the NRC Incident Investigation Team could review the licensee's proposed actions. The letter also stipulated that the plant was not to be restarted until authorized by the NRC Region III Regional Administrator or his designee.

Also on June 10, 1985, based on the potential safety implications of the event which were worthy of further study, the NRC Executive Director for Operations appointed a special NRC Incident Investigation Team in conformance with a NRC staff-proposed Incident Investigation Program. The Davis-Besse event was the first opportunity for the NRC to utilize this Program. The Team, composed of four technical experts, was to (1) fact-find as to what happened; (2) identify the probable cause as to why it happened; and (3) make appropriate findings and conclusions to form the basis for possible follow-on actions.

The Team began their investigation at the plant site on June 11, 1985. The equipment which malfunctioned was quarantined.

The Team collected and evaluated information to determine the sequence of operator, plant, and equipment responses during the event and the causes of equipment malfunctions. The sequence of these responses was determined primarily by interviewing personnel who were at the plant during the event and by reviewing plant data for the period immediately preceding and during the event. The Team also toured the plant to examine the equipment which malfunctioned, the equipment that was key to mitigating the transient, and the control room instrumentation and controls. The Team also interviewed plant management personnel and NRC Region III personnel who arrived at the site soon after the plant was stabilized about their knowledge of the plant response and operator actions. By correlating plant records with personnel statements on their actions and observations, the Team was able to compile a picture of the event.

The results of the Team's investigation are contained in NUREG-1154 (Ref. 7). Problems identified included issues specific to Davis-Besse and several possible generic issues. In addition, the Team made the major conclusion that the underlying cause of the loss of main and auxiliary feedwater event was the licensee's

lack of attention to detail in the care of plant equipment. The licensee has a history of performing troubleshooting, maintenance and testing of equipment, and of evaluating operating experience related to equipment in a superficial manner and, as a result, the root causes of problems are not always found and corrected. Engineering design and analysis effort to address equipment problems has frequently either not been utilized or has not been effective. Furthermore, operator interviews made clear that equipment problems were not aggressively addressed and resolved beyond compliance with NRC regulatory requirements.

An advance copy of NUREG-1154 was sent to the licensee on July 26, 1985. Pursuant to the provisions of 10 CFR § 50.54(f), the NRC sent a letter (Ref. 4) to the licensee on August 14, 1985, requiring the licensee to provide plans and programs to resolve numerous areas of concern identified by the NRC. The licensee responded on September 10, 1985 (Ref. 5). The response is being reviewed by the NRC.

The NRC Executive Director for Operations has assigned specific NRC Office responsibilities for resolution of the problem areas. In addition, general overall responsibilities include (1) the NRC Office of Nuclear Reactor Regulation for coordination of staff actions relating to restart of the plant, and (2) the NRC Region III Office for followup and reporting on the licensee's continued troubleshooting and determination of the root causes for the equipment failures. The NRC Region III Office also will perform plant inspections, as necessary, and determine possible enforcement actions.

The NRC continues to be involved in the resolution of this event and related matters. The event provides an opportunity for the NRC to learn from experience and to feed back the pertinent lessons into NRC and licensee activities. The Executive Director for Operations has directed NRC program managers to conduct an in-depth and searching reappraisal of the effectiveness of their programs in light of the lessons of the Davis-Besse event with the view of making the NRC programs more effective and the NRC a better regulatory agency. For example, what actions are needed when a utility continues to receive low overall performance ratings; what impediments or procedures are delaying decisions regarding plant upgrades; how can effective corrective action be achieved when plants have a history of maintenance deficiencies; and what should be done when voluntary licensee improvement programs prove less than satisfactory?

A specific example of the above pertaining to Davis-Besse is that the issue of upgrading the reliability of feedwater flow was first raised in 1979; however, at the time of the June 9, 1985 event, the issue remained unresolved.

The event generated considerable attention by the Commission, the ACRS, the media, the public and Congress. Congressman Edward J. Markey, Chairman, Subcommittee on Energy Conservation and Power, Committee on Energy and Commerce, U.S. House of Representatives, raised a number of issues regarding the event and the methods NRC uses to regulate licensees. A briefing was given to Congressman Markey and his staff. Formal responses have been, and will be, provided to questions as requested.

On July 8, 1985, Inspection and Enforcement Information Notice No. 85-50 was issued to inform licensees of this event (Ref. 8).

Future reports will be made as appropriate.

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FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the second calendar quarter of 1985. 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 second calendar quarter of 1985. As of the date of this report, the NRC had determined that the following events were abnormal occurrences.

85-8 Diagnostic 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 an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - On December 28, 1984, the NRC Region I office received written notification dated December 24, 1984, of a diagnostic misadministration that occurred at the Hospital of St. Raphael (the licensee), New Haven, Connecticut, described as having taken place on August 7, 1984. A patient was administered a 10 millicurie (mCi) dose of iodine-131 instead of an intended 400 microcurie (uCi) dose of iodine-123.

Nature and Probable Consequences - On July 30, 1984, a Nuclear Medicine Technologist received an order by telephone, from the secretary of a referring physician, requesting an iodine scan (or procedure) for one of the physician's patients. A capsule containing 10 mCi of iodine-131 was available. After the technologist received the approval of a resident physician, this capsule was administered orally to the patient at 1:00 p.m. on July 30, 1984. The nominal iodine-131 radioactivity level of the capsule dose was determined to be 15 mCi at 12:00 noon, CDT, July 27, 1984. The technologist was convinced that the patient was to have a diagnostic whole body scan as required in the protocol for the treatment and follow-up of patients with thyroid carcinoma.

The whole body scan of the patient was conducted on August 2, 1984, (rather than August 7, as initially reported by the licensee). The results of the whole body

scan performed on the patient revealed to the licensee that an error had been made. Lugol's solution was given to the patient in an attempt to minimize the effects of the iodine-131 uptake by the thyroid.

The original intent of the physician was to perform a diagnostic scan of the patient's thyroid gland using 400 uCi of iodine-123 in an attempt to determine whether excess tissue, apparently observed earlier in July 1984, was part of the thyroid (a goiter) or not. The earlier diagnostic scans were performed using 100 uCi of iodine-123 and 10 mCi of technetium-99m. An error was made in communicating and/or transcribing the July 30 request for a followup scan from the physician through to the technologist.

The licensee stated that in December 1984 the patient's hormone values were in the borderline hypothyroid level and the hospital recommended that the patient be placed on exogenous thyroid medication, which may have to be continued for life. The patient has been placed on this recommended medication and is being followed by the physician. Additional hormone studies will be performed by the hospital.

As a consequence of this misadministration, the hospital determined that the patient received in the order of 2,000 rads to the thyroid from the iodine-131. (NRC Region I personnel estimated an exposure of 8,000 rads to the thyroid based upon the "standard man" parameters). An NRC medical consultant has been asked to evaluate the difference between the two dose assessments.

Cause or Causes - Based on the inspection findings, the principal cause of this incident appeared to be inadequate communications between physicians and technologists. Other causal parameters were the lack of written orders or schedules; the lack of review, approval and scheduling of procedures through the radiologist; and the possible need for some clinical retraining of the technologists.

Actions Taken to Prevent Recurrence

Licensee - Clinical retraining has been provided for the technologist. Procedures have been established by which all doses, other than those that are routine diagnostic procedures, must be checked by the physician in charge of the clinic that day and not by a resident physician, and all radioiodine and special studies will be scheduled through the radiologist who must review the patient's charts and approve the specific study.

NRC - A special safety inspection was conducted in which this misadministration was reviewed. Several items of noncompliance with regulatory requirements were identified. An NRC medical consultant has been requested to assess the clinical aspects of the occurrence and verify or refute the appropriateness of the licensee's evaluations, dose assessment, and actions. Enforcement action is being reviewed by NRC Region 1.

On July 22, 1985, the NRC issued Inspection and Enforcement Information Notice No. 85-61 to all licensees authorized to use byproduct material for human applications, to inform them of this event, as well as of three other misadministrations to patients undergoing thyroid scans (Ref. 9).

Further reports will be made as appropriate.

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85-9 Diagnostic 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 an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - As described in a written notification dated April 1, 1985, to the NRC Region I Office, on March 19, 1985 a patient of Mercy Hospital (the licensee) in Pittsburgh, Pennsylvania, received 5 millicuries of iodine-131 rather than 10 millicuries of technetium-99m for a routine thyroid scan. The normal iodine-131 dose for thyroid uptake and scan is 5 to 100 microcuries.

Nature and Probable Consequences - On March 19, 1985, two out-patients were scheduled for nuclear scans at Mercy Hospital. One was scheduled for a routine thyroid scan using 10 millicuries of technetium-99m. This dose normally results in a dose to the thyroid of about 30 to 40 rads. The second patient was scheduled for a iodine-131 whole body scan. This study is less common and is normally done following the removal of a cancerous thyroid to detect metastases in other parts of the body. The physician had prescribed 5 millicuries of iodine-131 for this study. When the patient who was scheduled for the routine technetium-99m study arrived, the nuclear medicine technologist gave the patient the dose for the whole body scan without verifying the patient's identity. The mistake was recognized immediately afterwards and potassium perchlorate was promptly given to reduce the thyroid uptake.

The consequence of this incident was that a patient received an unnecessary dose to the thyroid estimated by the licensee to be about 1000 rads. The Nuclear Medicine physician believes this will result in no long or short term complications. An NRC medical consultant is reviewing the case.

Cause or Causes - A nuclear medicine technologist failed to verify identification of the patient prior to administration of a diagnostic dose.

Actions Taken to Prevent Recurrence

Licensee - Since 1980 the licensee has used an identification system for in-patients that requires both an oral query and check of the patient's wrist bracelet. Out-patients also are identified by oral query and by wrist bracelets which are assigned to out-patients at time of registration. The licensee has re-emphasized to its personnel the importance of correctly identifying patients before administration of doses.

NRC - No violations of NRC regulations were associated with this incident. However, as mentioned previously, the incident is being reviewed by an NRC medical consultant.

On July 22, 1985, the NRC issued Inspection and Enforcement Information Notice No. 85-61 to all licensees authorized to use byproduct material for human applications, to inform them of this event, as well as of three other misadministrations to patients undergoing thyroid scans (Ref. 9).

Further reports will be made as appropriate.

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85-10 Breakdown in Management Controls

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

Date and Place - On May 24, 1985, the NRC issued an Order (effective immediately) to Pittsburgh Testing Laboratory (PTL), Pittsburgh, Pennsylvania which required (1) the removal of the District Manager and Radiation Safety Officer (DM/RSO) for the licensee's Cleveland, Ohio facility, and (2) the suspension of all licensed activities at the Cleveland, Ohio facility until certain conditions were implemented (Ref. 10). This action was taken as a result of (1) assigning uncertified people to perform radiography, (2) providing false information to the NRC, and (3) falsifying training records.

Nature and Probable Consequences - During an NRC inspection conducted on August 27 and 29-31, 1984 of the operations at the licensee's facility in Cleveland, Ohio, and corporate office in Pittsburgh, Pennsylvania, violations of NRC requirements were identified which included the use of two uncertified personnel as industrial radiographers (Ref. 11).

The first uncertified individual was deliberately directed by the DM/RSO of the Cleveland, Ohio facility to perform the duties of a radiographer on seven occasions (i.e., on February 21 and 24, March 1, 2, 5, 7, and 9, 1984). Members of the corporate office staff subsequently became aware that the individual had acted as a radiographer prior to having been certified to perform radiographic operations, yet apparently took no action to inform corporate management of this violation or to take corrective action to prevent recurrence.

About five months later, the DM/RSO of the Cleveland, Ohio facility deliberately directed another uncertified individual to perform the duties of a radiographer on two occasions (i.e., on August 1 and 2, 1984), again knowing that the individual was not certified.

The investigation also showed that the DM/RSO had falsified the training records of the uncertified individual who performed licensed radiography activities during February and March 1984, so as to indicate that the individual had received the required training.

When questioned about the occurrences during the August inspection, the DM/RSO denied them. In a subsequent NRC investigation, he admitted the occurrences and admitted deliberate violation of NRC requirements.

These violations represent a continuing negative trend within the PTL organization involving inadequate control and implementation of radiation safety programs at PTL facilities throughout the country. In the past few years, violations have been identified at several PTL district offices by the NRC or Agreement States. A civil penalty was issued by the NRC in 1984 for violations at the Pittsburgh facility, and by the State of Tennessee in 1983 for violations at the Nashville facility. Following NRC issuance of the civil penalty in 1984, Mr. R. C. DeYoung, then Director of NRC's Office of Inspection and Enforcement (OIE), met with the licensee president on March 22, 1984 in Bethesda, Maryland, to discuss the NRC's concern regarding inadequate management control of licensed

activities. At that meeting, the licensee president acknowledged the need for improved corporate control of licensed activities to assure adherence to requirements at all PTL district facilities.

There is no evidence that any overexposures occurred while the uncertified personnel performed radiography. However, the use of radiographic devices by these personnel constituted a significant hazard not only to themselves, but also to their fellow workers and to several members of the public who were working in the areas where the radiography was being performed.

Cause or Causes - The root cause of the violations is attributed to a serious breakdown in management controls, both at the licensee's Cleveland facility and at the corporate headquarters in Pittsburgh.

Actions Taken to Prevent Recurrence

Licensee - To date, the licensee has complied with the requirements of the NRC Order described below.

NRC - On February 26, 1985, an enforcement conference was held with the licensee to discuss these violations, their causes and the licensee's corrective actions (Ref. 12). Based on this conference and the NRC investigation, the Director, OIE, determined that the deliberate actions of the Cleveland Manager in using uncertified individuals to perform radiography, and his subsequent lack of candor with NRC inspectors, demonstrated that there was no longer a reasonable assurance that the licensee would comply with Commission requirements while the Cleveland Manager was the Radiation Safety Officer at the Cleveland facility.

Accordingly, on May 24, 1985, the Director, OIE, determined that the public health and safety required issuance of an Order (effective immediately) requiring (1) the removal of the Manager from the position of Radiation Safety Officer of the Cleveland facility and from all involvement in the performance or supervision of NRC licensed activities and (2) the suspension of all licensed activities at the Cleveland facility until a qualified individual was appointed as Radiation Safety Officer of the Cleveland facility and authorized by the NRC (Ref. 10).

Further enforcement action is pending.

Further reports will be made as appropriate.

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85-11 Therapeutic Medical Misadministration

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

Date and Place - On June 17, 1985, Christ Hospital (the licensee), Cincinnati, Ohio, reported to the NRC that a 57-year old patient had received a 14,000 rad dose to the left lung instead of the prescribed 5,000 rad dose.

Nature and Probable Consequences - The patient was to be treated for lung cancer using implanted iridium-192 radiation sources to deliver a prescribed radiation dose of 5,000 rads to the left lung. In planning the treatment, the radiation physicist determined that 12 seeds (tiny sealed radiation sources), each containing 10 millicuries of iridium-192, would be sufficient for the treatment. The radiation physicist stated that she did not calculate the treatment time needed for delivering the prescribed dose, but the physician on the case stated that a 50-hour treatment period had been proposed by the physicist.

On June 17, 1985, following discussions between the physician and the radiation physicist, the radiation physicist realized that no calculations had been performed to determine the treatment period. The physicist then determined that 14,000 rad had been delivered to the left lung at that time, and the iridium seeds were promptly removed.

The patient is being monitored by the licensee through periodic examinations and bronchoscopy. No short term effects have been observed. The nature of the radiation therapy with the iridium-192 seeds is such that the principal radiation effects would be very localized in the left lung area. An NRC medical consultant has reviewed the case and is satisfied that adequate medical followup is being provided by the licensee.

Cause or Causes - The misadministration was caused by the failure of the radiation physicist to calculate the treatment period for the iridium-192 seeds to remain implanted in the patient. There is a discrepancy between the information supplied by the physician and the radiation physicist--the physician stating that a 50-hour period was proposed by the physicist, and the physicist stating that she had made no calculations for treatment time prior to the treatment and had not proposed a 50 hour treatment period.

The hospital had no quality assurance procedures designed to verify the accuracy of radiation treatment plans to assure that they would deliver the prescribed radiation dosage.

Actions Taken to Prevent Recurrence

Licensee - On June 19, 1985, the hospital implemented revised treatment procedures requiring that written calculations to determine a radiation therapy plan be performed prior to the initiation of a treatment and that the calculations be reviewed by a second qualified individual. The procedures were submitted to the NRC on June 28, 1985. The licensee's Medical Review Board approved the procedures on August 27, 1985.

NRC - The NRC conducted a special inspection of the licensee's radiation therapy program and the circumstances of the misadministration on June 26, 1985. The licensee's corrective actions were determined to be acceptable, and no violations of NRC requirements were identified in the inspection. Although no violations were identified, the NRC was concerned that appropriate corrective actions would be taken. Therefore, on September 11, 1985, the NRC issued a Confirmatory Order Modifying License (Ref. 13) requiring the licensee to immediately implement the revised treatment procedures.

This incident is closed for purposes of this report.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (See Appendix A) and report the events to the NRC for inclusion in this report. During the second calendar quarter of 1985, an Agreement State (Texas) reported the following abnormal occurrence to the NRC.

AS85-5 Overexposures of a Radiographer and an Assistant Radiographer

Appendix A (see Example 3 of "For All Licensees") of this report notes that exposure of the whole body of any individual to 25 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On February 6, 1985, a radiographer and an assistant radiographer employed by World Technical Services in Deer Park, Texas, received 8.3 rems and 34.3 rems whole body exposures, respectively. The exposures took place at the Amoco refinery in Texas City, Texas.

Nature and Probable Consequences - The radiographic crew was using a 100 curie iridium-192 radiographic source. Prior to the incident, the radiography crew, a radiographer, and assistant had radiographed pipe at another location at the Amoco refinery and had disconnected the equipment for transport to the new job site. When the crew arrived at the second job site, the radiographer attached film to the first weld to be radiographed and the assistant radiographer, who had been employed by the company for two weeks, assembled the radiographic device. This procedure included the connection of the source pigtail to the crank out and the connection of the guide tube to the radiographic device. The radiographer did not check to ensure that the connections were made correctly. After performing the first exposure, the exposure device was moved to a second position by the assistant radiographer. The film from the first radiograph was removed and new film was placed on the second weld by the radiographer. Between radiographs the radiographer and the assistant failed to observe the survey meter that had been left by the radiographic device or to survey the radiographic device or guide tube.

While approaching the radiographic device after the second radiograph, the radiographer noticed that the survey meter was off-scale after the source was supposed to be in its shielded position. The two employees withdrew to a safe distance and found their pocket dosimeters were off-scale. The radiographer then attempted to reduce the radiation levels by throwing film cassettes on the end of the source guide tube, where he believed the source was located.

When this failed to lower the radiation levels, the assistant radiographer was instructed to barricade as large an area as possible. After this was done the radiographer notified the company's radiation safety officer of the disconnect. The radiation safety officer in turn notified Gulf Nuclear, a company authorized to perform source retrieval.

Gulf Nuclear personnel proceeded to the location and were not able to dislodge the source from the guide tube. A storage container was obtained from Amoco and the source tube with the source was transferred to the Gulf Nuclear facility in Webster, Texas.

World Technical Services sent the employees' film badges to their personnel monitoring supplier for immediate processing. On the morning of February 7, 1985, the supplier notified the licensee that the radiographer received 8.3 rems whole body exposure and the assistant radiographer received 34.3 rems whole body exposure.

Cause or Causes - The cause of the disconnect was that the assistant radiographer was allowed to connect the source pigtail to the drive cable without the connection being checked by the radiographer. Causes of the exposures include failure to survey the radiographic device and guide tube between radiographs, the survey meter being left by the radiographic device while performing radiography, failure to check the survey meter when approaching the exposure device, and failure of the radiographer to follow the company's emergency procedures for a source disconnect.

Actions Taken to Prevent Recurrence

Licensee - The personnel have been reinstructed in the company's emergency procedures and the radiographers have been reinstructed to check the connections made by the assistant radiographers.

State Agency - An investigation of the incident was conducted by the Agency on February 8, 1985. Interviews of the employees were conducted and statements taken. Information concerning the sequence of the events leading to the source disconnect and exposures was consistent throughout the statements. A time/dose study of the incident was performed and photographs were obtained of both employees' hands.

Calculations were performed by the Agency to determine the possible location of the source in the guide tube and the exposure to the employees. From the calculations it appears that the greatest estimated exposure received was 85.5 rems to the waist of the assistant radiographer. The radiographer and assistant radiographer have been seen by a doctor knowledgeable in radiation injuries. From the doctor's report and the Agency's investigation, it appears that neither of the employees grasped the guide tube where the source was lodged.

When the source has decayed sufficiently, the Agency plans to have Gulf Nuclear open the source tube so that the location of the source can be determined and the rear portion of the pigtail can be checked.

The Agency cited the company for failure to survey the radiographic device and guide tube, failure to follow the company's operating and emergency procedures which prohibit leaving the survey meter by the camera during radiographs, and failure to follow the company's safety procedures for a source disconnect.

This incident is closed for purposes of this report.

REFERENCES

1. Letter from S. J. Collins, Chief, Projects Branch No. 2, Division of Project and Resident Programs, NRC Region I, to J. D. O'Toole, Vice President, Nuclear Engineering and Quality Assurance, Consolidated Edison Company of New York, Inc., forwarding Inspection Report No. 50-247/84-33, Docket No. 50-247, February 14, 1985.*

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

3. Letter from Stephen M. Quennoz, Plant Manager, Davis-Besse Nuclear Power Station, to U.S. Nuclear Regulatory Commission, Document Control Desk, forwarding Licensee Event Report No. 85-13, Docket No. 50-346, July 9, 1985.*

4. 10 CFR § 50.54(f) letter from Harold R. Denton, Director, NRC Office of Nuclear Reactor Regulation, to Joe Williams, Jr., Senior Vice President, Nuclear, Toledo Edison Company, Docket No. 50-346, August 14, 1985.*

5. Letter from John P. Williamson, Chairman and Chief Executive Officer, Toledo Edison Company, to Harold R. Denton, Director, NRC Office of Nuclear Reactor Regulation, Docket No. 50-346, September 10, 1985.*

6. 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, June 10, 1985.*

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

8. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-50, "Complete Loss of Main and Auxiliary Feedwater at a PWR Designed by Babcock & Wilcox," July 8, 1985.*

9. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-61, "Misadministrations to Patients Undergoing Thyroid Scans," July 22, 1985.*

10. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to M. Ruyan, President, Pittsburgh Testing Laboratory, forwarding an Order to Show Cause Why License Should Not be Suspended and Modified (Immediately Effective), Docket No. 30-05985, May 24, 1985.*

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

**Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for inspection. Available for purchase from the GPO Sales Program, Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7982.

11. Letter from Thomas T. Martin, Director, Division of Radiation Safety and Safeguards, NRC Region I, to M. Y. Ruyan, President, Pittsburgh Testing Laboratory, forwarding Inspection Report No. 30-05985/84-02, Docket No. 30-05985, February 8, 1985.*

12. Letter from Thomas T. Martin, Director, Division of Radiation Safety and Safeguards, NRC Region I, to M. Ruyan, President, Pittsburgh Testing Laboratory, forwarding Enforcement Conference Report No. 30-05985/85-01 of an Enforcement Conference held on February 26, 1985, Docket No. 30-05985, April 19, 1985.*

13. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to Jack Cook, President, The Christ Hospital, forwarding a Confirmatory Order Modifying License (effective immediately), License No. 34-03831-02, September 11, 1985.*

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

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

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

An event will be considered an abnormal occurrence if it involves a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

1. Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission;
2. Major degradation of essential safety-related equipment; or
3. Major deficiencies in design, construction, use of, or management controls for licensed facilities or material.

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

For All Licensees

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual to 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation (10 CFR §20.403(a)(1)), or equivalent exposures from internal sources.
2. An exposure to an individual in an unrestricted area such that the whole-body dose received exceeds 0.5 rem in one calendar year (10 CFR §20.105(a)).
3. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceed 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR §20.403(b)).
4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit.
5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.
6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

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

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

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

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

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

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

For Commercial Nuclear Power Plants

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

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

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

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

5. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity 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 Licensees

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

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

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

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the April through June, 1985 period, the NRC, NRC licensees, Agreement States, Agreement States Licensees, and other involved parties, such as reactor vendors and architects and engineers, continued with the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. The referenced Congressional abnormal occurrence reports below provide the initial and any updating information on the abnormal occurrences discussed. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

76-11 Steam Generator Problems

This abnormal occurrence was originally reported in NUREG-0090-5, "Report to Congress on Abnormal Occurrences: July - September 1976," under the title of "Steam Generator Tube Integrity," and updated in subsequent reports in the series, i.e, NUREG-0090-8; Vol. 1, No. 4; Vol. 2, No. 3; Vol. 2, No. 4; Vol. 3, No. 1; Vol. 3, No. 2; Vol. 3, No. 4; Vol. 4, No. 1; Vol. 5, No. 2; Vol. 7, No. 2; and Vol. 7, No. 3. In Vol. 5, No. 2, the title was changed to "Steam Generator Problems" since the scope of the reporting was expanded to include more than steam generator tube problems. The item is further updated as follows.

In 1982, Indian Point Station Unit 3 had a leak at weld No. 6 on one of their steam generators. Weld No. 6 is a full penetration circumferential weld located in the transition zone between the tube bundle and steam dryer areas, below the feedwater nozzles, and subject to thermal cycling. The crack was started by corrosion and operating temperature fluctuations caused it to grow through the wall because of low cycle fatigue. The repair method reduced the defects to an acceptable level. Ultrasonic examinations have been performed during outages since 1982; the most recent of which was conducted in the summer of 1985. Previously known indications that appear to have grown in size are presently being evaluated.

In 1983, Surry Power Station Unit 2 performed ultrasonic examinations of the No. 6 welds. The original construction weld at Unit 2 is 6 inches above the weld that attached the lower portion of all three replacement steam generators in 1980. The examination showed widespread indications of discontinuities on the inside surface of this weld in the "A" steam generator. Additional magnetic particle testing revealed the existence of closely spaced linear cracks over a large portion of the circumference.

The cracks in generator A were in a narrow band at the upper edge of the weld and covered almost the entire inside diameter. The cracks were as deep as 0.5 inch and were covered by the surface oxide, which obscured detection by visual examination. Generators B and C had numerous smaller circumferential cracks in the same location. To complicate matters, there were 10 unacceptable sub-surface indications in generator B, based on the requirements of ASME Section XI, IWB-3511. After a fracture and fatigue evaluation, these sub-surface

indications were accepted by ASME IWB-3600. The surface cracks in all three generators were removed by grinding; repair welding was not necessary.

At the next outage, the No. 6 welds in all three steam generators at Surry Unit 2 will be partially examined by magnetic particle testing. The sub-surface indications in generator B also will be examined by ultrasonic methods.

The NRC staff believes the steam generator shell cracking at Indian Point and Surry are the result of thermally induced fatigue in the presence of corrosion over a long period of time. This is a generic problem, and therefore similar cracking may occur at other PWR facilities as their steam generators age. Inspection and Enforcement Information Notice No. 85-65, "Crack Growth in Steam Generator Girth Welds," was issued to pressurized water reactor licensees on July 31, 1985, to inform them of the Indian Point and Surry experiences (Ref. B-1).

The staff has recently concluded that the cracking phenomena does not represent an immediate generic safety concern because detectable through-wall crack leakage, in a manner similar to Indian Point, would be expected to precede large scale rupture of the steam generator shell.

This item is generally considered closed for purposes of this report. However, it occasionally is reopened to report steam generator information considered significant.

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77-9 Environmental Qualification of Safety-Related Electrical Equipment Inside Containment

This abnormal occurrence was originally reported in NUREG-0090-10, "Report to Congress on Abnormal Occurrences: October - December 1977," and updated in subsequent reports in this series, i.e., NUREG-0090; Vol. 1, No. 1; Vol. 1, No. 2; Vol. 2, No. 2; Vol. 3, No. 2; Vol. 4, No. 2; Vol. 5, No. 2; and Vol. 6, No. 1. It is further updated as follows.

A new rule, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," Section 50.49 to 10 CFR Part 50, was published in the Federal Register on January 21, 1983 and became effective on February 22, 1983 (Ref. B-2). The scope of the rule covers electrical equipment important to safety as defined by the rule. This includes electrical equipment important to safety both inside and outside containment that would be subject to harsh environment following design basis events. The rule specifies the qualification parameters and methods that an equipment qualification program should include. In addition, the rule specifies a deadline by which all electrical equipment covered by the scope of the rule should be qualified. This deadline is November 30, 1985.

Licensees have been notified by Generic Letter 85-15 (Ref. B-3) that extensions beyond the November 30, 1985 deadline are not envisioned; future action on environmental qualification is expected to be in the enforcement realm. Enforcement guidelines for environmental qualification deficiencies are now being developed.

Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," was issued in June 1984 (Ref. B-4). The Regulatory Guide clarifies the new rule and specifies acceptable methods for compliance with the rule.

In late 1984, the NRC Office of Inspection and Enforcement (OIE) implemented a program to inspect licensee compliance with 10 CFR §50.49. This program is designed to examine the adequacy and effectiveness of licensee procedures for establishing and maintaining qualification of equipment covered by the rule, including the review of qualification document files; verification of environmental qualification master list; surveillance and maintenance procedures and their use; as well as procurement of replacement parts and equipment. Physical inspection of installed equipment is also included in the scope of this program.

Based on the findings of the first phase of the inspection program, OIE issued Information Notice No. 85-39, "Auditability of Electrical Equipment Qualification Records at Licensees' Facilities," on May 22, 1985 to alert licensees of certain deficiencies identified in these inspections (Ref. B-5).

Concurrently with the licensee inspections, OIE is also conducting a program of periodic inspections of suppliers of environmental qualification (EQ) services (vendors) including test laboratories, system suppliers, and equipment manufacturers. These inspections are designed to examine, on a sample basis, the interfaces between the licensees and organizations supplying qualified equipment or performing EQ services. One objective of the inspections is to verify that the applicable requirements of 10 CFR §50.49 are passed on to the vendors supplying EQ services and that these services are in fact performed in accordance with the specified requirements. As a result of these inspections, Information Notice No. 85-40, "Deficiencies in Equipment Qualification Testing and Certification Process," was issued on May 22, 1985 which identified observed deficiencies in the equipment qualification documentation, review, approval and certification (Ref. B-6).

Unless new significant information become available, this item is closed for purposes of this report.

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79-3 Nuclear Accident at Three Mile Island

This abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and updated in each subsequent report in this series, i.e., NUREG-0090, Vol. 2, No. 2 through Vol. 8, No. 1. It is further updated as follows:

Reactor Building Entries

During the second calendar quarter of 1985, 69 entries were made into containment. There have been a total of 643 entries since the March 28, 1979 accident. Reactor building activities during this period centered on preparations for both reactor vessel plenum assembly (PA) transfer and early defueling, including the installation of a dam in the fuel transfer canal, partial installation of the defueling water cleanup system (DWCS) and fuel transfer equipment, load testing of the 25-ton polar crane auxiliary hoist and annual preventative maintenance on the polar crane. The PA was successfully transferred and stored during this period, as discussed below.

Plenum Assembly Transfer

On Wednesday, May 15, 1985, the PA was successfully lifted from its jacked position in the reactor vessel, transported horizontally above the refueling canal, and lowered into its storage stand in the deep end of the canal, where it is currently shielded under five feet of water. A dam was erected in the shallow end of the canal prior to PA removal to allow flooding of the deep end to a level that will provide adequate shielding of the stored PA throughout defueling. The actual PA transfer and storage resulted in an occupational exposure of approximately 3 person-rem, well below the range of 25-50 person-rems previously estimated for the activity.

EPICOR-II/Submerged Demineralizer System (SDS) Processing

The EPICOR-II system processed approximately 144,006 gallons of water during the reporting period. The SDS processed approximately 148,575 gallons of water during that time.

Liner Shipments

The nineteenth SDS vessel and three spent resin liners were sent from the site to the DOE facility at Hanford, Washington, during this reporting period. A dewatered EPICOR-II resin liner was sent to Richland, Washington.

TMI-2 Advisory Panel Meetings

The Advisory Panel for the Decontamination of Three Mile Island Unit 2 (Panel) met on April 11, 1985, in Lancaster, Pennsylvania. The Panel received presentations from the licensee on distribution of fuel in the primary system and plenum removal. The NRC staff gave a presentation on the staff's review of the potential for inadvertent recriticality. The licensee and the NRC staff provided information on radiation protection issues related to the TMI-2 cleanup.

At the May 16, 1985 meeting, the Panel discussed their position on the level of the Panel's inquiry into health effect studies and data related to the radioactive release during the TMI-2 accident. The Panel received a presentation from representative of General Public Utilities Nuclear Corporation (GPUNC) on plans for reactor fuel removal and storage. The Department of Energy briefed the Panel on the current status of fuel shipping casks that will be used for offsite transport of fuel and debris removed from the reactor. The NRC staff provided the Panel with an update on the status of NRC investigations and enforcement actions.

On June 20, 1985, Panel members met with the NRC Commissioners in Washington, DC. Topics for discussion included recent reactor vessel plenum removal, fuel shipping casks, worker radiation protection plan, and the recent GPUNC revision to the cleanup schedule. Also discussed were issues of information flow from the NRC to the Panel and the Panel's involvement in health studies related to the TMI-2 accident. The NRC staff provided an update on enforcement actions and investigations related to the cleanup.

Further reports will be made as appropriate.

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83-6 Uncontrolled Leakage of Reactor Coolant Outside Primary Containment

NUREG-0090, Vol. 6, No. 3, "Report to Congress on Abnormal Occurrences: July - September 1983," described an August 25, 1982 event at Hatch Unit 2 which involved a sustained and uncontrolled loss of hot pressurized reactor coolant outside the primary containment.

The following event has certain similarities to the Hatch Unit 2 event. The event occurred on July 12, 1985 at Oyster Creek Unit 1 while the plant was at 100% power. Oyster Creek Unit 1 utilizes a General Electric-designed boiling water reactor. The plant is operated by GPU Nuclear Corporation and is located in Ocean County, New Jersey.

At 9:38 a.m. with the plant operating normally at full power and a reactor coolant system pressure of 1020 psig, a mechanical failure of the plant's electric pressure regulator (EPR) caused a turbine bypass valve to fully open. This resulted in a decrease in reactor pressure which caused an automatic closure of the main steam isolation valves (MSIVs). As designed, the reactor protection system initiated an automatic insertion (scram) of the control rods.

Although all rods inserted to shut down the reactor, one (the south volume) of the two scram discharge volumes (SDVs) did not fully isolate: one drain valve (V-15-121) failed to shut fully and the redundant drain valve (V-15-134) reopened slightly as valve V-15-121 leakage allowed pressure under the seat of valve V-15-134 to increase.

As reactor coolant leaked past the control rod drive (CRD) seals and into the SDV and reactor building equipment drain tank (RBEDT), the SDV and associated piping began to heat up. Also, hot water being drained into the RBEDT flashed to steam. The paint on the SDV began to blister and fume. Steam in the RBEDT emanated from the hub drains in the reactor building, particularly around the south SDV (directly above the RBEDT). The combination of the steam and fumes eventually caused activation of a portion of the deluge fire system . Water wetted down safety-related instrument racks and electrical and mechanical equipment in the southwest areas of the 51' and 23' elevations. The licensee declared an Unusual Event at 10:35 a.m. because of the potential loss of safety equipment from the deluge system activation.

The SDVs (north and south) are used to limit the loss of and contain reactor vessel water from all the control rod drives during a scram. The failure of the south SDV to properly isolate after the reactor scram represented a degradation of primary containment in that it allowed reactor vessel water to pass through the SDV into the reactor building environment. This failure, coupled with activation of the deluge system, resulted in elevated airborne radiation levels in the reactor building as well as potential operability problems for safety related systems due to high humidity conditions. In addition, the event caused the CRD seals to be subjected to excessive temperatures.

These conditions would continue to exist until the scram could be reset by sending closure signals to the scram inlet and outlet valves on each CRD. By design, the scram could not be reset until reactor pressure was reduced to less than 600 psig. This could be done by initiation of the isolation condensers (to begin plant cooldown), which would automatically occur when the increasing reactor pressure (due to the closure of the MSIVs) reached the initiation set point.

However, problems were encountered which delayed initiation of the isolation condensers. First, reactor water level had been allowed to rise beyond the level instrument span. The high water level, and the inability to determine how high the level was, delayed use of the isolation condensers (ICs) due to operator concern for possible water hammer damage to the IC piping and pipe supports. The reactor operators immediately overrode the impending automatic initiation of the IC on high reactor pressure and sought ways to control reactor water level and pressure.

As reactor pressure continued to increase, two electromatic relief valves (EMRVs) opened when their set point was reached. The reactor operators then took manual control of the EMRVs, per procedure, and controlled pressure between 950 and 1000 psig. The containment spray system was started in the torus cooling mode. At this point, the reactor water level had decreased from steaming to within the range of control room level instrumentation and pressure control, using the "B" IC, was initiated.

When pressure was reduced to less than 600 psig, the scram was reset which closed the scram inlet and outlet valves, terminating the release of hot reactor coolant and steam that was entering the reactor building. During this evolution of lowering pressure, reactor water level shrunk to below the reactor low water level setpoint and a second scram signal was received. Water level was promptly restored via the feedwater system and the scram was reset again. Following this scram reset, the reactor operators placed the plant into a cold shutdown condition. The total lapsed time from the initial reactor scram until the final scram reset was approximately 40 minutes.

After plant conditions were stable and all safety-related equipment potentially affected by the deluge system was verified operable, the licensee terminated the Unusual Event declaration at 1:22 p.m. on June 12, 1985.

Following declaration of an Unusual Event at 10:35 a.m., the licensee implemented emergency preparedness procedures to monitor potential offsite releases. No detectable releases were noted. The licensee also implemented environmental monitoring by taking soil and water samples immediately below the plant vents, and by taking smear samples of the plant access road. No radioactivity attributed to the event was detected.

In addition, the licensee closely monitored radiation levels and potential inplant airborne contamination during the event in order to minimize possible personnel exposures.

The event was initiated by the failure of the EPR. The EPR servo valve was stuck in a position which called for the turbine control valves and one bypass valve to open. The cause of the stuck valve was attributed to impurities in the hydraulic oil.

Some of the delay in initiating the isolation condensers just after the scram was attributed to a reactor operator action problem regarding control of reactor vessel water level.

Failures of the south SDV drain valves, V-15-121 and V-15-134, to fully close were attributed to improper stroke adjustment and a design deficiency, respectively. It was found that the V-15-121 valve stroke was 1/8" short of being fully closed.

(The resulting through leakage caused pressurization of the piping between valve V-15-121 and downstream valve V-15-134.) The actuator closing spring for V-15-134 was found to be undersized, resulting in the valve opening slightly when pressure, due to V-15-121 leakage, was applied under its seat.

A reactor water cleanup (RWCU) system isolation valve did not initially open due to a trip of its supply breaker. The breaker tripped because the Limitorque operator for the valve did not develop sufficient torque to move the valve off its seat. This insufficient torque was a result of improper gear ratios in the operator.

A contributing cause to the delay in terminating the small LOCA was the present plant design in that the scram signal can not be bypassed or reset until reactor pressure decreases below 600 psig, independent of the position of the reactor mode selector switch. Until the scram can be reset, the SDV vent and drain valves act as containment isolation valves.

Following the event, plant management conducted a post-reactor trip review meeting to determine the causes of the event and to plan appropriate corrective actions. The causes of the equipment failures were determined as discussed above. Immediate routine actions included:

1. The sticking EPR control valve was rebuilt, filters for the valve were replaced, and the oil tubing supplying the valve was flushed. Results of subsequent testing of the EPR were satisfactory.

2. The stroke was adjusted on SDV valve V-15-121. In addition, all other SDV vent and drain valves were checked for proper stroke adjustment. No deficiencies were noted.

The actuator spring for SDV valve V-15-134 was replaced with a properly sized spring for the valve's particular application. The valve performed satisfactorily during subsequent pressure testing.

3. The gear ratio was changed in the RWCU isolation valve Limitorque operator. The valve performed satisfactorily during subsequent testing.

4. As discussed previously, all equipment potentially affected by the activation of the deluge system was verified operable. In addition, the scram times of a selected number of CRDs were measured and found to be acceptable, indicating that the CRD seals were unaffected by the event.

Following the actions above, the licensee restarted the plant on June 18, 1985.

Additional corrective actions planned, or underway, by the licensee include:

1. Discuss the event during operator training.

2. Conduct an investigation to determine why the incorrect spring was installed in the actuator for south SDV valve V-15-134

3. Review the present EPR filtration system for possible improvement (to prevent impurities in the hydraulic control oil). Alternative control valve designs, which would eliminate this type of control valve failure, will also be evaluated.

4. In addition, the licensee is reviewing the present design which does not permit bypassing the scram signal until reactor pressure decreases below 600 psig to determine whether any changes should be considered.

The NRC resident inspector was in the control room for most of the scram recovery period and observed the response of the operators and plant equipment to the event.

Following the event, a radiation specialist for NRC Region I went to the plant to review the licensee's responses during the event regarding emergency preparedness, environmental monitoring, and radiation protection of plant personnel.

In addition, the circumstances associated with the event were reviewed as part of the routine resident safety inspections performed from June 3-30, 1985, of licensee operations.

Within the scope of the inspections performed, no violations were identified. The results of the inspections were forwarded to the licensee on July 26, 1985 (Ref. B-7).

On August 22, 1985, the NRC issued Inspection and Enforcement Information Notice No. 85-72 to all boiling water reactor plants having an operating license or a construction permit to inform them of this event (Ref. B-8).

The event remains under review by the NRC. However, unless new significant information becomes available, the incident is considered closed for purposes of this report.

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84-8 Degraded Isolation Valves in Emergency Core Cooling Systems

This abnormal occurrence was originally reported and closed out in NUREG-0090, Vol. 7, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1984." It is being reopened to report the following new information.

As stated in the previous report, several events have occurred which involved open valves, including check valves (valves designed to allow water to flow only in one direction), located in the emergency core cooling systems of various General Electric-designed, boiling water reactors. Some of the events resulted in the high pressure reactor coolant system overpressurizing piping in either the low pressure coolant injection (LPCI) subsystem of the residual heat removal (RHR) system, the high pressure coolant injection (HPCI) system (the low pressure suction portion), or the low pressure core spray system; each of these systems are designed to help mitigate the consequences of a loss of coolant accident (LOCA). These events are considered to be significant because they substantially reduced safety margins for preventing an interface LOCA.

The previous report mentioned that the NRC Office for Analysis and Evaluation of Operational Data (AEOD) was performing a more detailed study on this subject. During this study, two additional events (which were not described in NUREG-0090, Vol 7, No. 3) were found. Both events occurred at LaSalle Unit 1 and both involved the same valves. These events are described below:

1. On October 5, 1982, with the plant operating at 20% power, quarterly surveillance testing on the high pressure core spray (HPCS) system was being conducted. The testable isolation check valve, and its associated bypass valve, failed to indicate completely closed after they were opened for the test. Both the testable isolation check valve and its bypass valve are situated on the HPCS injection line inside primary containment. The HPCS system was declared inoperable and was isolated by deactivating the normally closed motor-operated HPCS injection valve.

During the surveillance test, the bypass valve was first opened to equalize the pressure on both sides of the testable check valve disk. The testable check valve was then tested open by operating a remote handswitch. This handswitch energized a solenoid valve to allow instrument air to be supplied to one side of the piston cylinder of the air operator of the testable check valve, causing the piston cylinder to move a rack and gear assembly against spring tension. The rack and gear assembly movement rotated the actuator rod which lifted the valve disk off its seat. When the handswitch was returned to its closed position, the solenoid valve was de-energized, cutting off instrument air supply to the piston cylinder. This should have allowed the spring (tension) to return the rack and gear assembly to its normal position. This, in turn, should have rotated the actuator rod back to its original position, allowing the valve disk to reclose by its own weight and differential pressure.

The failure of the testable check valve to reclose was investigated by the licensee and was determined to have been caused by (a) dried lubricant on the actuator piston cylinder, (b) insufficient preload on the actuator spring assembly, and (c) the stuck open bypass valve. Together, these causes prevented the piston cylinder of the check valve air operator from returning to its fully retracted position.

2. On June 17, 1983, with the plant at 48% power and quarterly operating surveillance of the HPCS system in progress, the HPCS testable isolation check valve, and its associated bypass valve, failed to indicate closed after being tested open. The HPCS system was declared inoperable and was isolated by deactivating the normally closed motor-operated HPCS injection valve.

The licensee determined that the failure of the testable isolation check valve to reclose was caused by (a) the stuck open bypass valve which prevented a pressure differential from developing across the valve disk of the testable check valve, and (b) possible thermal binding of the testable check valve disk. With respect to the latter cause, the licensee indicated that the Anchor Darling check valve and bypass valve have a tendency if tested hot to remain partially open after being cycled. The failure of the bypass valve to reclose was traced to insufficient return spring tension in the bypass valve. While shutting down the plant, both the bypass valve and the testable check valve closed without any assistance as reactor pressure and temperature decreased. Subsequent to an analysis of the event, the licensee submitted a request to conduct surveillance testing of the testable check valve only during cold shutdown.

AEOD performed a generic review and evaluation of these two events, in addition to the six events previously described in NUREG-0090, Vol. 7, No. 3. The AEOD report (Ref. B-9) states that for the eight events (each of which involved the failure of a testable isolation check valve on the injection line of an emergency core cooling system), five involved an additional failure of the second and

final isolation barrier -- the inadvertent opening of a normally closed motor-operated injection valve. Four of these five events occurred during power operation, thus leading to an actual overpressurization of an emergency core cooling system. Each of these operational events is considered a precursor to an interfacing loss-of-coolant accident between the reactor coolant system and an emergency core cooling system. Such an accident would involve the sudden discharge of reactor coolant at operating pressure and temperature outside the primary containment and would also likely disable one or more of the safety systems required to mitigate the accident.

Collectively, these operating events have serious safety implications (i.e., the likelihood of an interfacing loss-of-coolant accident is higher by two to several orders of magnitude than had been previously assessed). This data provides a strong indication that prompt corrective actions should be taken to prevent a recurrence of these reported multiple failures.

AEOD developed several recommendations (both short and long term) for consideration by the responsible NRC lead office, to eliminate the root cause(s) of these occurrences. The recommendations appear to involve minimum costs, and if effectively implemented should significantly reduce the reactor accident risks associated with such multiple failures. AEOD will track the recommendations to assure that they are satisfactorily resolved, after the report is issued.

This incident is closed for purposes of this report.

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84-9 Degraded Shutdown Systems

This abnormal occurrence was originally reported in NUREG-0090, Vol. 7, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1984," and updated in NUREG-0090, Vol. 7, No. 4. It is further updated as follows.

Background

As previously reported, Fort St. Vrain (FSV) tripped (scrammed) from low power on June 23, 1984 and six out of the total 37 control rod pairs failed to insert upon receipt of the automatic scram signal from the plant protective system. Consequently, the plant was shut down for refurbishment of the control rod drives and extensive maintenance. During this period, several issues, in addition to the refurbishment of the control rod drives, were identified which required resolution before the plant could resume operations. These additional issues were, (1) the licensee reported during July 1984 that numerous control rod position instrumentation anomalies had occurred; (2) during testing of two hoppers of the reserve shutdown system on November 5, 1984, one hopper failed to release all of its contained graphite balls; and (3) a special assessment by the NRC of the licensee's (Public Service Company of Colorado) overall operation found significant weaknesses in every area of the operation that was audited. These issues were resolved as described below.

Control Rod Drive Mechanism (CRDM) Refurbishment

Because of the stress corrosion cracking of the 347 stainless steel suspension cables associated with the CRDMs, the licensee decided to completely refurbish

each of the 37 CRDMs installed in the reactor. This work began in early February 1985, and was completed in late June 1985. It involved complete disassembly of each mechanism, thorough cleaning of all parts, installation of new Inconel-625 suspension cables, and replacement of all bearings in the mechanisms with new bearings.

Following reassembly, each of the CRDMs was tested to verify operability and to obtain base line data for in-service testing to detect any future deterioration in operating performance.

Because of the importance of the licensee's CRDM refurbishment program, NRC carried out an intensified inspection effort involving NRC inspectors and a specialist in the design, fabrication, and maintenance of precision mechanisms obtained from the Idaho National Engineering Laboratory. The overhaul work was witnessed at each essential point; because of this intense NRC inspection scrutiny, it was concluded with high confidence that each of the refurbished mechanisms now installed in the reactor has been restored to good operating condition.

Control Rod Position Instrumentation

The control rod position instrumentation was repaired and/or replaced as necessary, and tested to assure satisfactory operation. The licensee is developing some long range plans to improve the instrumentation.

Reserve Shutdown System

The boron carbide balls in the reserve shutdown system hoppers in each of the 37 control rod drive assemblies were replaced with new material having a very low residual B_2O_3 impurity level. This was done to prevent the boron carbide balls from sticking together should moisture leach the impurity out of the balls.

NRC Special Assessment of the Licensee's Overall Operations

The special assessment was carried out as a joint effort by NRC's licensing and inspection staffs with technical assistance from the Los Alamos Scientific Laboratory. The results of this assessment were sent to the licensee on October 16, 1984. The licensee has responded to each of the issues raised in the assessment report, and the NRC staff has concluded that the actions taken by the licensee in resolving these issues are acceptable. These include the CRDM refurbishment work discussed above, as well as the development of special procedures for in-service testing and surveillance of the CRDMs. Still ongoing as a result of the NRC assessment is work of a longer range nature to which the licensee is committed. This work includes development of a preventive maintenance program for the CRDMs, upgrading the licensee's technical specifications, implementing improvements in management structure and practices related to plant operations, and conducting various engineering studies leading to possible modifications to plant equipment to improve performance.

Authorization for Resumption of Operations

As stated previously, inspection activity was intensified during the CRDM refurbishment program. In addition, because of the long duration of the outage, NRC also carried out special inspections during June and July 1985, to verify readiness of the plant and the operations staff to resume power operations. It was

determined that all repair work had been satisfactorily completed, system configurations were appropriate, training of plant personnel was current, and the facility was ready to be operated.

On July 19, 1985, the NRC Regional Administrator of Region IV authorized the licensee to resume reactor operations (Ref. B-10) at a power level no greater than 15% rated thermal power pending resolution of some technical questions concerning equipment qualification. (These questions are not related to this abnormal occurrence.) Full power operations will resume when the questions concerning equipment qualification are satisfactorily resolved.

The letter authorizing resumption of operations also summarized binding licensee commitments and schedules concerning completion of long-term work identified in the NRC special assessment report and various ongoing technical activities.

This incident is closed for purposes of this report.

APPENDIX C

OTHER EVENTS OF INTEREST

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

1.0 Deficiencies in Quality Assurance Program During Construction

The Waterford Steam Electric Station, Unit 3 (WSES-3) utilizes a Combustion Engineering-designed pressurized water reactor and is located on the Mississippi River, 25 miles west of New Orleans in St. Charles Parish. It is operated by the Louisiana Power and Light Company (LP&L). On May 24, 1985, the NRC Regional Administrator of Region IV transmitted to LP&L a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $130,000 as a result of weaknesses identified in the implementation of LP&L's quality assurance (QA) program during construction of WSES-3 (Ref. C-1).

Previously, in mid-1983, construction activities at WSES-3 were nearing completion when NRC began receiving numerous allegations about the adequacy of construction at WSES-3. Those broad allegations, which led to this enforcement action, claimed, in general, that a breakdown in the QA program at WSES-3 had occurred. Consequently, NRC's Executive Director for Operations issued a directive on March 12, 1984, to centralize responsibility for management of all items needing resolution prior to the licensing decision in order to assure that all relevant issues were properly addressed. This action included the formation of the Waterford Task Force which was composed of more than 40 technical specialists from headquarters and regional offices, and included several NRC consultants. An early Task Force decision was to concentrate on the timely identification and resolution of technical safety issues and to delay formal enforcement actions until the conclusion of the Task Force effort.

The Waterford Task Force findings were documented in Supplement No. 7 and Supplement No. 9 of the Safety Evaluation Report (NUREG-0787), which relates to the operation of WSES-3 (Ref. C-2). The overall conclusion of the Task Force was that "Although LP&L did not fully implement their QA program and there was a partial breakdown involving some subcontractors, the impact on hardware was minimal. The plant design is adequate and the final installed plant systems meet the design requirements. The few hardware deficiencies were reviewed and were found to have been satisfactorily corrected."

At the conclusion of the Task Force effort, potential violations were processed by NRC Region IV. These violations had been determined from several concurrent and related efforts involving approximately 20,000 staff hours associated with the NRC Inquiry Team, the Construction Appraisal Team (CAT), the Waterford Task Force, and routine NRC Region IV inspections. As an overview, these violations included a failure to make adequate corrective actions; a failure to assure qualifications of QA personnel; a failure to adequately maintain, control, or process safety-related records; and a failure to properly implement various aspects of the QA program.

The NRC Region IV Regional Administrator noted in his May 24, 1985, letter (Ref. C-1) that, although the violations did not appear to have led to an end-product of unacceptable quality, they illustrated weaknesses that existed in LP&L's implementation of its QA program during construction and that the violations were of concern to NRC because LP&L's responsibility for QA does not end with receipt of an operating license.

The LP&L response to NRC's Notice of Violation and Proposed Imposition of Civil Penalties is currently being evaluated by the NRC Region IV staff.

REFERENCES
FOR APPENDICES

B-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-65, "Crack Growth in Steam Generator Girth Welds," July 31, 1985.*

B-2 U.S. Nuclear Regulatory Commission, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," Federal Register Vol. 48, No. 15, January 21, 1983, 2729-2734.

B-3 Generic Letter 85-15, "Information Relating to the Deadlines for Compliance with 10 CFR 50.49 (Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants)," August 6, 1985.*

B-4 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1, issued June 1984.**

B-5 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-39, "Auditability of Electrical Equipment Qualification Records at Licensees' Facilities," May 22, 1985.*

B-6 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-40, "Deficiencies in Equipment Qualification Testing and Certification Process," May 22, 1985.*

B-7 Letter from Harry B. Kister, Chief, Projects Branch No. 1, Division of Reactor Projects, NRC Region I, to P. B. Fiedler, Vice President and Director, Oyster Creek Nuclear Generating Station, forwarding Inspection Report No. 50-219/85-19, Docket No. 50-219, July 26, 1985.*

B-8 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-72, "Uncontrolled Leakage of Reactor Coolant Outside Containment," August 22, 1985.*

B-9 U.S. Nuclear Regulatory Commission, Case Study Report, "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors," prepared by Peter Lam, NRC Office for Analysis and Evaluation of Operational Data. Preliminary report, dated February 1985 was issued March 7, 1985.* Final report (designated as AEOD/C502) was issued in September 1985.*

B-10 Letter from Robert D. Martin, Regional Administrator, NRC Region IV, to O. R. Lee, Vice President, Electric Production, Public Service Company of Colorado, Docket No. 50-267, July 19, 1985.*

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

**Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for inspection. Available for purchase from the GPO Sales Program, Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7982.

C-1 Letter from Robert D. Martin, Regional Administrator, NRC Region IV, to R. S. Leddick, Senior Vice President-Nuclear Operations, Louisiana Power and Light Company, forwarding a Notice of Violation and Proposed Imposition of Civil Penalty, Docket No. 50-382, May 24, 1985.*

C-2 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station Unit No. 3, Docket No 50-382," USNRC Report NUREG-0787, issued July 1981.** Pertinent supplements issued as follows:

Supplement No. 7, issued September 1984.**
Supplement No. 9, issued December 1984.**

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

**Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for inspection. Available for purchase from the GPO Sales Program, Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7982.

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Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period April 1 to June 30, 1985. During the report period, there were three abnormal occurrences at the nuclear power plants licensed to operate. These events involved, respectively, (1) inoperable safety injection pumps, (2) significant deficiencies in reactor operator training and material false statements, and (3) loss of main and, auxiliary feedwater systems. There were four abnormal occurrences at other NRC licensees. Three events involved diagnostic or therapeutic medical misadministrations; the other involved a breakdown in management controls. There was one abnormal occurrence reported by an Agreement State; the event involved overexposures of a radiographer and an assistant radiographer. The report also contains information updating some previously reported abnormal occurrences.

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