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PREVIOUS REPORTS IN THE SERIES

Report to the Congress on Abnormal Occurrences, January-June 1975, USNRC Report, NUREG 75/090, October 1975.

Report to Congress on Abnormal Occurrences, July-September 1975, USNRC Report, NUREG-0090-1, March 1976.

Report to Congress on Abnormal Occurrences, October-December 1975, USNRC Report, NUREG-0090-2, March 1976.

Report to Congress on Abnormal Occurrences, January-March 1976, USNRC Report, NUREG-0090-3, July 1976.

Report to Congress on Abnormal Occurrences, April-June 1976, USNRC Report, NUREG-0090-4, October 1976.

Report to Congress on Abnormal Occurrences, July-September 1976, USNRC Report, NUREG-0090-5, March 1977.

Report to Congress on Abnormal Occurrences, October-December 1976, USNRC Report, NUREG-0090-6, June 1977.

Report to Congress on Abnormal Occurrences, January-March 1977, USNRC Report, NUREG-0090-7, June 1977.

Report to Congress on Abnormal Occurrences, April-June 1977, USNRC Report, NUREG-0090-8, September 1977.

Report to Congress on Abnormal Occurrences, July-September 1977, USNRC Report, NUREG-0090-9, November 1977.

Report to Congress on Abnormal Occurrences, October-December 1977, USNRC Report, NUREG-0090-10, March 1978.

Report to Congress on Abnormal Occurrences, January-March 1978, USNRC Report, NUREG-0090, Vol. 1, No. 1, June 1978.

Report to Congress on Abnormal Occurrences, April-June 1978, USNRC Report, NUREG-0090, Vol. 1, No. 2, September 1978.

Report to Congress on Abnormal Occurrences, July-September 1978, USNRC Report, NUREG-0090, Vol. 1, No. 3, December 1978.

Report to Congress on Abnormal Occurrences, October-December 1978, USNRC Report, NUREG-0090, Vol. 1, No. 4, March 1979.

Report to Congress on Abnormal Occurrences, January-March 1979, USNRC Report, NUREG-0090, Vol. 2, No. 1, July 1979.

Report to Congress on Abnormal Occurrences, April-June 1979, USNRC Report, NUREG-0090, Vol. 2, No. 2, November 1979.

Report to Congress on Abnormal Occurrences, July-September 1979, USNRC Report, NUREG-0090, Vol. 2, No. 3, February 1980.

Report to Congress on Abnormal Occurrences, October-December 1979, USNRC Report, NUREG-0090, Vol. 2, No. 4, April 1980.

Report to Congress on Abnormal Occurrences, January-March 1980, USNRC Report, NUREG-0090, Vol. 3, No. 1, September 1980.

Report to Congress on Abnormal Occurrences, April-June 1980, USNRC Report, NUREG-0090, Vol. 3, No. 2, November 1980.

Report to Congress on Abnormal Occurrences, July-September 1980, USNRC Report, NUREG-0090, Vol. 3, No. 3, February 1981.

ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from October 1 to December 31, 1980

The following incidents or events, including any submitted by the Agreement States, were determined by the Commission to be significant and reportable:

1. There was one abnormal occurrence at the nuclear power plants licensed to operate. The event involved significant flooding of a reactor containment building.

2. There were no abnormal occurrences at the fuel cycle facilities (other than nuclear power plants).

3. There were no abnormal occurrences at other licensee facilities.

4. There were no abnormal occurrences reported by the Agreement States.

This 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 (42 FR 10950) on February 24, 1977 (Ref. 1). In order to provide wide dissemination of information to the public, a Federal Register notice is issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. At a minimum, each such notice contains the date and place of the occurrence and describes its nature and probable consequences.

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period between October 1 to December 31, 1980.

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

THE REGULATORY SYSTEM

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

In the licensing and regulation of nuclear power plants, the NRC follows the philosophy that the health and safety of the public are best assured through the establishment of multiple levels of protection. These multiple levels can be achieved and maintained through regulations which specify requirements which will assure the safe use of nuclear materials. The regulations include design and quality assurance criteria appropriate for the various activities licensed by NRC. An inspection and enforcement program helps assure compliance with the regulations. Requirements for reporting incidents or events exist which help identify deficiencies early and aid in assuring that corrective action is taken to prevent their recurrence.

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

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

REPORTABLE OCCURRENCES

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

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

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by NRC to the nuclear industry, the public, and other interested groups as these events occur. Dissemination includes deposit of incident reports in the NRC's public document rooms, special notifications to licensees and other affected or interested groups, and public announcements. In addition, a biweekly computer printout containing information on reportable events received from NRC licensees is sent to the NRC's more than 120 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

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

AGREEMENT STATES

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

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

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

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

OCTOBER-DECEMBER 1980

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth quarter of 1980. As of the date of this report, the NRC had determined that the following was an abnormal occurrence. The abnormal occurrence was determined reportable during preparation of this report and therefore has not previously been noticed in the Federal Register. Federal Register noticing is being made in conjunction with the noticing of issuance of this report.

80-9 Significant Flooding of Reactor Containment Building

Appendix A (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. In addition, Appendix A (Example 4 of "For Commercial Nuclear Power Plants") of this report notes that discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or Technical Specifications that requires immediate remedial action can be considered an abnormal occurrence.

(As described below, the licensee took exception to certain of the NRC findings, including the issues pertaining to management controls.)

Date and Place - On October 17, 1980, a significant amount of water was discovered inside the containment building at Consolidated Edison Company's Indian Point Unit 2 facility. Indian Point Unit 2 utilizes a pressurized water reactor and is located in Westchester County, New York.

Nature and Probable Consequences - Upon entry into the Unit 2 containment building on October 17, 1980, to repair a malfunctioning power range nuclear detector, a significant amount of water was discovered on the containment floor, in the containment sumps, and in the cavity under the reactor pressure vessel. The amount of water which had accumulated was later determined to be about 125,000 gallons. The source of the water was found to be service water from leaks in service water piping and from leaks in the containment fan cooling units. Failures and inadequacies of components and systems required to remove and detect water in the containment building resulted in the accumulation of such a large quantity of water without detection.

The flooding directly resulted in the failure of a power range nuclear detector; its repair was the original reason for containment building entry. Because of the flooding, the cavity under the reactor vessel was nearly filled, resulting in the wetting of the lower 9 feet of the reactor vessel and submergence of stainless steel conduits and instrument thimbles located below the reactor vessel.

Although the direct consequences of the event were not significant, the accumulation of the large amount of water in containment and the wetting of the reactor vessel raised significant safety questions. Evaluations to date indicate that there was no damage to the reactor vessel or other components in the reactor vessel cavity; however, continued operation with abnormal conditions that were not known (the undetected accumulation of water in the containment) did represent some degree of decreased safety.

Cause or Causes - This event resulted from the following combination of conditions: (1) there were significant multiple service water leaks from the containment fan cooling units onto the containment floor. These coolers have a history of such leakage and they cannot be detected by inventory losses since the service water system is an open system supplied from a river; (2) both containment sump pumps were inoperable, one due to blown fuses and the other due to binding of its controlling float switch; (3) a series of containment sump level indicating lights for indicating increasing water level had shown no change for an extended period of time; (4) there was no high water level alarm to indicate the overflowing sump level; (5) the moisture level indicators for the containment atmosphere did not indicate high moisture levels, apparently due to an error in calibration and/or ranging which made them insensitive to the moisture levels resulting from relatively small cold water leaks; (6) the two submersible pumps in the cavity under the reactor pressure vessel were ineffective since they pump into the containment sump (and the sump pumps were inoperable); (7) there was no water level instrumentation in the cavity under the reactor vessel; (8) there was no indication outside containment that the pumps in the cavity under the reactor vessel were operating; and (9) the holdup tanks which ultimately receive water pumped from the containment sump also receive water from other water sources such as Unit 1 process water, lab drain water, etc., and these other water sources masked the effect of cessation of water flow from the Unit 2 containment sump.

The result of an investigation into the event conducted by the NRC's Office of Inspection and Enforcement indicated that deficiencies in the licensee's management system directly contributed to the event. The licensee's failure to identify and correct the causes of leakage, to require routine containment inspections, or to establish adequate controls to insure that systems required to remove water from the containment were operable, led directly to the flooding event. The investigation also showed that the facility was restarted on October 20, 1980, without adequate evaluation of the potential consequences of the event with regard to continued plant operation. The NRC investigation also concluded that certain plant modifications had not been properly evaluated and that the NRC had not been promptly informed of the flooding incident.

Actions Taken to Prevent Recurrence

Licensee - The licensee has taken the following actions: (1) installed alarms in the control room indicating increasing containment sump levels; (2) installed alarms in the control room to indicate when either submersible pump in the reactor cavity operates; (3) repaired the service water leaks; (4) installed guide bushings on the containment sump pump control floats to prevent their binding; and (5) repaired the containment sump water level indicators. The licensee also plans to replace the containment fan unit cooling coils prior to return to power from the current refueling outage. Further actions are also being evaluated in response to the NRC letters described below.

NRC - On October 22, 1980, the NRC Region I office issued an immediate action letter to the licensee confirming the licensee's commitments to specific actions to prevent recurrence prior to restart of the plant.

The NRC staff determined that the event demonstrated a serious weakness in the licensee's management control system. As a result, on December 11, 1980, the staff proposed imposition of civil penalties in the amount of $210,000 for violations, associated with the event, including failure to promptly report the event (Ref. 2). The NRC also identified four potential unreviewed safety questions associated with the event. The licensee responded by letters dated January 5 and February 11, 1981. The licensee contested some aspects of the proposed penalty and the conclusion on serious weakness in the management control system. The licensee also responded to the potential unreviewed safety questions. After review of the licensee's response, the staff concluded that there was no basis for mitigation of the civil penalties. Accordingly, on March 2, 1981, an Order imposing Civil Penalties in the amount of $210,000 was sent to the licensee (Ref. 3). Further responses in regard to corrective actions were also requested from the licensee. The licensee's response to the potential unreviewed safety questions is still under review and will be the subject of separate NRC correspondence.

IE Information Notice 80-37 was issued on October 24, 1980 (Ref. 4) to all holders of operating licenses and construction permits to provide them with the details of this occurrence. On November 21, 1980, IE Bulletin No. 80-24 (Ref. 5) was issued directing all licensees at operating plants to take specific short-term actions and to report information to the NRC. Licensees with plant designs similar to Indian Point Unit 2 were directed to verify or provide specific equipment and procedural controls to preclude events similar to that which occurred at Indian Point Unit 2. NRC will evaluate the reports submitted by all licensees to determine what other generic longer-term actions may be required.

Further reports will be made as appropriate.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth quarter of 1980. As of the date of this report, the NRC had not determined that any were abnormal occurrences.

OTHER NRC LICENSEES

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

There are currently more than 8,500 NRC nuclear material licenses in effect in the United States, principally for use of radioisotopes in the medical, industrial, and academic fields. Incidents were reported in this category from licensees such as radiographers, medical institutions, and byproduct material users.

The NRC is reviewing events reported by these licensees during the fourth quarter of 1980. As of the date of this report, the NRC had not determined that any were abnormal occurrences.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth quarter of 1980, the Agreement States reported no abnormal occurrences to the NRC.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence Reports: Implementation of Section 208, Energy Reorganization Act of 1974; Policy Statement," Federal Register, Vol. 42, No. 37, February 24, 1977, 10950-10952.

2. Letter from Victor Stello, Jr., NRC, to Arthur Hauspurg, Consolidated Edison Company of New York, Inc., forwarding a Notice of Violation and Proposed Imposition of Civil Penalty and a Notice of Deviation, Docket No. 50-247, December 11, 1980.*

3. Letter from Victor Stello, Jr., NRC, to Arthur Hauspurg, Consolidated Edison Company of New York, Inc., forwarding an Order Imposing Civil Monetary Penalties, Docket No. 50-247, March 2, 1981.*

4. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 80-37, "Containment Cooler Leaks and Reactor Cavity Flooding at Indian Point Unit 2," October 24, 1980.*

5. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment (October 17, 1980 Indian Point 2 Event)," November 21, 1980.*

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

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

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

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

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

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

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

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

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 Part 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 Part 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 Part 20.403(b)).

4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit (10 CFR Part 71.36(a)).

5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.

6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

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

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

9. An accidental criticality (10 CFR Part 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 Part 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 systems).

For Fuel Cycle Licensees

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

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

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

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the October through December 1980 period, the NRC, NRC licensees, Agreement States, Agreement State licensees, and other involved parties, such as reactor vendors and architects and engineers, continued with the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. The referenced Congressional abnormal occurrence reports below provide the initial and any updating information on the abnormal occurrences discussed. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

The following abnormal occurrence was originally reported in NUREG-75/090, "Report to Congress on Abnormal Occurrences: January - June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1, 2, 3, 9, Vol. 1, No. 3, Vol. 2, No. 2, Vol. 2, No. 4, and Vol. 3, No. 2. It is further updated as follows:

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

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in boiling water reactors (BWRs) since the mid-1960s. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure mode by being "sensitized," either by welding or by post-weld heat treatment. Although the likelihood is extremely low that IGSCC-induced cracks will propagate far enough to create a significant hazard to the public, the occurrence of such cracks is undesirable, and measures to minimize IGSCC in BWR piping systems are indicated to improve overall plant reliability.

A final "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" (NUREG-0313, Revision 1) was issued in July 1980 (Ref. B-1). The report sets forth the NRC staff's revised guidelines for reducing IGSCC susceptibility of BWR piping. The guidelines describe a number of preventive and corrective measures acceptable to the NRC, including guidelines for: (1) corrosion-resistant metals for installation in BWR piping, (2) methods of testing. (3) processing techniques, (4) augmented inservice inspection, and (5) leak detection. The report also included recommendations for developmental work to provide future improvements in limiting the extent of IGSCC or detecting it when it occurs. The report also includes public comments which were evaluated, and several modifications to the report were made to accommodate those comments.

This constitutes the completion of the generic technical activity A-42. The staff is now in the process of implementing the position established in the above-mentioned NUREG-0313, Revision 1 report.

Since a technical resolution of this problem has been completed, no further updates on this problem are planned for future issues of these quarterly abnormal occurrence reports to Congress (NUREG-0090 series). Progress on implementation will be reported annually in the section of the NRC's Annual Report (Ref. B-2) which addresses Unresolved Safety Issues, and quarterly in NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" (Ref. B-3).

This incident is closed for purposes of this report.

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The following abnormal occurrence was originally reported in NUREG-0090-3, "Report to Congress on Abnormal Occurrences: January - March 1976," and updated in subsequent reports in this series, i.e., NUREG-0090-4, 6, Vol. 1, No. 1, Vol. 1, No. 3, and Vol. 2, No. 3. It is furter updated as follows:

76-1 Deficiencies in the Mark I Containment Systems of Certain Boiling Water Reactors (BWRs)

The NRC staff has identified and initiated two separate generic tasks to review and evaluate the results of the industry programs and to develop criteria for licensing actions on individual plants using Mark I containment designs. Task A-6 was completed with the issuance of the "Mark I Containment Short-Term Program Safety Evaluation Report" (NUREG-0408, December 1977 - Ref. B-4). In that report, the NRC concluded that an adequate margin of safety had been demonstrated for the most probable hydrodynamic loads induced by a design-basis LOCA, such that the licensed Mark I BWR facilities may continue operation without undue risk to the health and safety of the public while the Long-Term Program is being conducted.

Task A-7 was concluded with the issuance of the "Mark I Containment Long-Term Program Safety Evaluation Report" (NUREG-0661, July 1980 - Ref. B-5). This report describes the results of the NRC's review of the proposed generic hydrodynamic load definition and structural assessment techniques and the NRC Acceptance Criteria for the subsequent plant-unique assessments. The plant-unique assessments are currently underway and most of the affected utilities have performed several of the known plant modifications in order to expedite the resolution of this issue. The Acceptance Criteria, together with schedules for completion of all of the plant modifications needed to conform to these criteria, were formally issued on January 13, 1981 to the Mark I licensees. The completion schedules for modifying the Mark I containment systems in accordance with the Long-Term Program range from October 1981 to January 1983. The schedules were developed in consideration of the types of modifications needed, the availability of technicians and materials, and the schedules for plant refueling outages.

Since a technical resolution of this problem has been completed, no further updates on the problem are planned for future issues of these quarterly abnormal occurrence reports to Congress (NUREG-0090 series). Progress on implementation will be reported annually in the section of the NRC's Annual Report (Ref. B-2) which addresses Unresolved Safety Issues, and quarterly in NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" (Ref. B-3).

This incident is closed for purposes of this report.

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The following abnormal occurrence was originally reported in NUREG-0090-5, "Report to Congress on Abnormal Occurrences: July-September 1976," 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, and Vol. 3, No. 2. It is further updated as follows:

76-11 Steam Generator Tube Integrity

The NRC staff continues to closely monitor, review and evaluate, and approve the acceptability of continued operation of plants experiencing steam generator tube problems. Plant technical specifications require routine inservice inspection of steam generators to be performed every 12 to 24 months. The NRC has imposed license conditions on plants with severely degraded steam generators to increase the required frequency of inspection. The conditions also require that, following inspection of steam generators and completion of any necessary repair programs by the licensees, the NRC must approve or concur in the restart of each severely affected facility. Safe operation is assured by the imposition of strict conditions, including the plugging of affected tubes and restricting of allowable leak rates during operation.

Work is proceeding with three generic tasks in the NRC program for the resolution of generic issues. Specifically involved are Generic Tasks A-3, A-4, and A-5, addressed to the problems of Westinghouse, Combustion Engineering, and Babcock and Wilcox steam generators, respectively.

The approach taken in these tasks is to integrate technical studies in the three areas of systems analyses, inservice inspection, and tube integrity in order to establish improved criteria by which to ensure safe and reliable steam generator operation. These studies have been completed and a draft report will be issued for public comment in April 1981.

Further reporting on the progress of the generic studies is provided annually in the section of the NRC's Annual Report (Ref. B-2) which addresses Unresolved Safety Issues, and quarterly in NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" (Ref. B-3).

Further reports on unique operating experience or problems will be made as appropriate in these quarterly abnormal occurrence reports to Congress (NUREG-0090 series). For the present, this incident is closed for purposes of this report.

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The following abnormal occurrence was originally reported in NUREG-0090-6, "Report to Congress on Abnormal Occurrences: October-December 1976," and updated in subsequent reports in this series, i.e., NUREG-0090-7, NUREG-0090, Vol. 1, No. 4, and Vol. 2, No. 2. It is further updated as follows:

76-16 Feedwater Nozzle Cracking in Boiling Water Reactors (BWRs)

Resolution of this problem has been pursued by the NRC staff under the Generic Task A-10 since 1977. In November 1980, the staff issued its final edition of NUREG-0619, entitled "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" (Ref. B-6). This document provides the staff's resolution of this generic issue. It describes the technical issues, the technical studies and analyses performed by the General Electric Company and the NRC staff, the staff's technical positions based on these studies, and the staff's requirements for continued implementation of its technical positions. Public comments received in response to the "For Comment" edition of April 1980 have been incorporated where applicable. Appendix E of NUREG-0619 discusses the comments and the staff's disposition of these comments.

The NRC staff has concluded, in the case of the feedwater nozzles, that the combination of nozzle clad removal, installation of triple-sleeve spargers designed by General Electric (or others with satisfactory characteristics), procedural changes, and systems changes where deemed necessary, will assure the goal of long-term operation without significant crack growth. However, no operating reactor currently satisfies all requirements. The staff has established inservice inspection intervals based upon its evaluation of the combined proposed solutions. The staff has also determined that licensees with interference-fit single sleeve spargers (the "interim" repair) may be allowed to continue operating until June 30, 1982, when required modifications are to be completed.

With regard to the control rod drive return line nozzle, the staff has concluded that certain control rod drive hydraulic system modifications will be necessary on all operating reactors. The staff's principal considerations were the need to prevent cracking of the nozzle and the need to increase the likelihood of high pressure control rod drive system flow to the reactor vessel sufficient to cover the core when other sources of water are unavailable. This was the case at some times during the March 1975 fire at Browns Ferry Unit No. 1 (Ref. B-7). The staff requires that each operating plant prove this capability, which may require the simultaneous operation of the two control rod drive pumps. This requirement, plus others contained in Part II of NUREG-0619, represents a conservative departure from current licensing criteria.

Plants currently under licensing review will be required to conform to the criteria developed for operating plants for both the feedwater nozzle and control rod drive return line nozzle issues.

Implementation of the technical resolution in NUREG-0619 was initiated with a letter to licensees dated November 13, 1980 which requests that the requirements contained in the NUREG be met in accordance with an established schedule.

Since a technical resolution of this problem has been completed, no further updates on this problem are planned for future issues of these quarterly abnormal occurrence reports to Congress (NUREG-0090 series). Progress on implementation will be reported annually in the section of the NRC's Annual Report (Ref. B-2) which addresses Unresolved Safety Issues, and quarterly in NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" (Ref. B-3).

This incident is closed for purposes of this report.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 1, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1978," and updated in subsequent reports in the series, i.e., NUREG-0090, Vol. 1, No. 4, and NUREG-0090, Vol. 2, No. 2. It is further updated as follows:

78-2 Fuel Assembly Control Rod Guide Tube Integrity (A Generic Concern)

As previously reported, unexpected wear of rodded guide tubes has been observed in discharged pressurized water reactor (PWR) fuel assemblies. It was concluded that coolant turbulence is responsible for inducing vibratory motion in the normally fully withdrawn control rods and, when these vibrating rods are in contact with the inner surface of the guide tubes, a fretting wear of the guide tube wall takes place. Significant wear is limited to the relatively soft Zircaloy-4 guide tubes because the Inconel-625 control rod cladding is a relatively wear-resistant material. The extent of the wear is both time- and plant-dependent and has in some cases extended completely through the guide tube wall.

Guide tubes function principally as the main structural members of the fuel assembly and as channels to guide and decelerate control rod motion. Significant loss of mechanical integrity due to wear or hole formation could (a) result in the inability of the guide tubes to withstand loadings during operational transients and fuel handling accidents and (b) hinder the ability to scram the control rods. Therefore, the NRC has been reviewing this issue on a generic basis with vendors and owners' groups and on a case-by-case basis with individual applicants and licensees.

Because of effective hardware modifications and satisfactory confirmatory surveillance that are discussed below, guide tube wear is considered to be under effective control; therefore, unless unforeseen problems arise, this item will not be further reported in these reports to Congress.

Combustion Engineering

Combustion Engineering (CE) continues to rely on chrome-plated, stainless-steel sleeves as guide tube inserts that effectively eliminate wear. But in pursuit of a permanent solution, CE has placed flow-modifying test assemblies in

Maine Yankee; Arkansas Nuclear One, Unit 2; Millstone Unit 2; and Calvert Cliffs Units 1 and 2. In 1979 and 1980, surveillance on guide tube integrity was reported on Fort Calhoun; Millstone Unit 2; St. Lucie Unit 1; and Calvert Cliffs Units 1 and 2.

The surveillance at Calvert Cliffs Unit 1 and Millstone Unit 2 revealed loose guide tube sleeves, which were subsequently repaired by recrimping. With the exception of these anomalies, sleeves continue to be an acceptable means of precluding guide tube wear. However, it is uncertain as to whether surveillance underway will find that the inclusion of sleeves has now increased the rate of wear on the Inconel control rods and thereby reduced their design lifetime.

Westinghouse

Westinghouse has submitted (a) guide tube wear measurements taken on 14x14 fuel assemblies from Point Beach Units 1 and 2, (b) a mechanistic wear model, and (c) the impact of the model's wear predictions on the safety analyses of plant designs.

Westinghouse believes that their fuel designs will experience less wear than those reported in some other plant vendor designs because the Westinghouse designs use thinner, more flexible control rods that have relatively more lateral support in the guide tube assembly of the upper core structure.

The NRC accepted the Westinghouse analyses for plants using the 14x14 and 15x15 fuel assembly designs. Because of the uncertainties in predicting wear for the 17x17 fuel assembly design, several near-term operating license applicants were required to submit a surveillance program. For acceptability, the minimum objective of such a program was to demonstrate that there is no occurrence of hole formation in rodded guide tubes. These applicants have formed an owners' group which is planning to submit the results of a surveillance program during 1981.

In 1980 Westinghouse provided reload fuel for the CE-designed plant Millstone Unit 2. Guide tube sleeves similar in concept to those previously used in that core were installed in all of the reload fuel except in a few flow-modifying demonstration assemblies. Confirmation on the adequacy of the sleeve and flow-modifying designs will be obtained at the next reload outage.

Babcock & Wilcox

Babcock & Wilcox (B&W) has assessed the potential for guide tube wear in their plants. The B&W assessment did not provide a means for predicting the rate of wear nor actual wear measurements on fuel assemblies; therefore, a near-term operating license applicant and all B&W licensees were required to provide confirmatory measurements on irradiated fuel assemblies that verify that the B&W designs will not experience through-the-wall wear in guide tubes.

Consequently, an owners' group was formed and submitted the details of a surveillance program conducted in Oconee Units 1 and 3 and Rancho Seco. The

NRC will evaluate this submittal, which states that the measured wear was low and within the design criteria.

Exxon

Exxon (ENC) has supplied the reload fuel for the CE-designed plant Maine Yankee. To prevent wear, the ENC fuel incorporated a sleeved guide tube design that is similar to that of the residual CE fuel assemblies.

In light of the lower wear rates in Westinghouse-designed plants, the ENC reload fuel for these designs does not employ modifications to preclude guide tube wear. In 1980 ENC performed surveillance on fuel assemblies that were discharged from the Westinghouse-designed plant H. B. Robinson Unit 2. Although no quantitative conclusions were determined, the measured wear did not extend through the guide tube walls.

This incident is closed for purposes of this report.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and updated in subsequent reports in this series, i.e., NUREG-0090 Vol. 2, No. 2, Vol. 2, No. 3, Vol. 2, No. 4, Vol.3, No. 1, Vol. 3, No. 2, and Vol. 3, No. 3. It is further updated as follows:

79-3 Nuclear Accident at Three Mile Island

Reactor Building Entries

Three reactor building entries (the third, fourth, and fifth since the March 28, 1979 accident) were made during the fourth calendar quarter of 1980. Current plans are to conduct reactor building entries at the rate of approximately one per month for the next several months.

The third reactor building entry was made on October 16, 1980. The entry team of three health physics technicians and two maintenance men remained in the reactor building for up to 87 minutes. The duration of the entry was extended past the scheduled 60 minutes to complete the assigned tasks which required more time than anticipated. The entry team was not exhausted physically and no entry team member exceeded the precalculated dose of 500 mr total body exposure. The entry team's physical comfort was improved by a decrease in protective clothing requirements and by cooler temperatures in the reactor building.

The entry team performed several tasks during this third entry. They took photographs, made radiation surveys, reset the tripped differential pressure interlock on the equipment hatch personnel airlock, removed an inoperable noise monitor preamplifier for offsite analysis, and removed the preamplifier for an inoperable neutron source range monitor. This preamplifier will be repaired (or replaced) and reinstalled during a subsequent entry.

The fourth reactor building entry was made on November 13, 1980. Three separate entries were made by a total of 12 men. No entry team member exceeded 500 mr total body exposure. A member of the onsite NRC staff was part of the entry team. The maximum stay time inside the reactor building for any individual was 90 minutes. The entry tasks included motion picture filming of the 305-foot and 347-foot elevations. The motion picture film, from inside the reactor building, was made available to the news media. Still photographs of the steam generators and the pressurizer were made. Several decontamination techniques were tried on test areas on the 305-foot elevation floor.

The fifth entry into the Unit 2 reactor building occurred on December 11, 1980. NRC's Deputy Program Director, TMI Program Office, was one of the fourteen men who participated in the entry. The maximum total body exposure for any member of the entry team was approximately 650 mr. The maximum exposure was accumulated during a radiation survey of "hot spots" on the 305 foot elevation. The ambient radiation levels on the 305-foot elevation were between 400 and 600 mr/hr. "Hot spots" on this elevation reached 20 R/hr. Other tasks completed included a survey of the reactor head from inside the refueling pool. The ambient gamma readings in the vicinity of the reactor head were 400 mr/hr. The maximum gamma radiation detected during the head survey was 800 mr/hr. Radiation levels between 100 and 200 mr/hr were recorded during a climb to the polar crane. The crane cab was not entered; however, visual observations from the platform immediately below the cab revealed considerable heat damage to electrical components.

Decontamination tests were performed during this fifth reactor building entry. The tests consisted of applying various commercial decontamination solutions and heated water under pressure to painted concrete and steel surfaces on the 347 foot elevation in the reactor building. Preliminary analyses of surface radioactivity before and after the application of the decontamination solutions indicate cleanup factors on the order of about two to ten were achieved for the removal of isotopes of cesium and about 10 to 20 for isotopes of strontium. Additional decontamination tests are planned for future reactor building entries.

Reactor Cooling

The licensee initiated (with NRC staff approval) a Loss of Decay Heat to Ambient Test on November 6, 1980 to determine if the TMI-2 reactor could be adequately cooled by transferring the reactor decay heat to the reactor building ambient. For this test, the turbine bypass isolation valve was closed, thereby terminating use of the condenser as a heat sink. Key plant temperatures (average value of the incore thermocouples and hottest loop hot leg temperature) decreased approximately 6°F during this test due to a decrease in ambient reactor building temperatures. The test was completed on December 9, 1980. At the completion of this test, the turbine bypass isolation valve was reopened and the reactor coolant mode was returned to cyclic natural circulation with heat rejection to the condenser.

The licensee has submitted a proposal justifying the loss to ambient mode as a viable means of decay heat removal based on test data. The TMI Program Office staff is reviewing the test data and the licensee's proposed procedures for utilizing this cooling mode.

On November 14, 1980, the NRC staff approved use of the newly installed Mini Decay Heat Removal System (MDHRS). The MDHRS provides an appropriately sized forced flow system for removing decay heat from the TMI-2 reactor fuel. Use of the MDHRS will simplify plant operations by eliminating the need for operating various systems which are required in the current cooling mode (heat rejection to the condenser by subatmospheric boiling in the "A" steam generator).

Draft Programmatic Environmental Impact Statement (PEIS)

The public comment period for "Draft Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979, Accident Three Mile Island Nuclear Station, Unit 2" (NUREG-0683 - Ref.B-8) expired on November 20, 1980. During the public comment period, the NRC staff participated in 31 meetings with the public, local officials, and organizations to receive comments on the PEIS. Verbatim transcripts were made of the major meetings and 151 relevant comments were received during the public meetings. In addition, 121 letters containing approximately 765 additional comments were received. The NRC staff is presently preparing the final PEIS which is scheduled for submittal to the Commission in late February 1981.

Advisory Panel

The Commission's newly established (October 1980) Advisory Panel for the Decontamination of TMI Unit 2 held its first three meetings November 12, December 18, and December 30, 1980. The Panel will consult with and provide advice to the Nuclear Regulatory Commission on major activities required to accomplish expeditious and safe cleanup of the TMI-2 facility. Members of the Panel include Chairman John E. Minnich, Dauphin County (PA) Commission; Dr. H. Arnold Muller, M.D., Secretary of the Pennsylvania Department of Health; Mr. Clifford Jones, Secretary of the Pennsylvania Department of Environmental Resources; Lt. Gen. DeWitt C. Smith, Jr., Director of the Pennsylvania Emergency Management Agency; Mayor Robert Reid of Middletown, PA; Mayor Arthur Morris of Lancaster, PA; Dr. Nunzio J. Palladino, Dean of Engineering at Pennsylvania State University; Dr. Henry J. Wagner, Jr., M.D., Professor of Medicine, Radiology and Environmental Health and Head of the Division of Nuclear Medicine and Radiation Health, the Johns Hopkins University, Baltimore, MD; Dr. Thomas Cochran, physicist, Natural Resources Defense Council, Washington, DC; Ann Trunk, Middletown, PA, member of the President's Commission on Three Mile Island; Joel Roth, Elizabethville, PA, Chairman of the TMI Alert organization; and Jean Kohr, Millersville, PA, attorney for the Susquehanna Valley Alliance.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 3, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1980." It is further updated as follows:

80-6 Failure of Control Rods to Insert Fully During a Scram

Continuous Monitoring System (CMS)

As reported in NUREG-0090 Vol. 3, No. 2, on October 2, 1980, the NRC issued Confirmatory Orders to the licensees of 16 Boiling Water Reactor (BWR) plants requiring the installation of equipment to continuously monitor water levels in all Scram Discharge Volumes (SDVs) and provisions for water level indication and alarm for each SDV in the control room. Until the system is installed and operating satisfactorily, the licensees shall increase their surveillance of the SDV water level. The equipment provides information to the reactor operator such that if water accumulates in the SDV, reactor operators may take timely action. This equipment was to be operable by December 1, 1980 or prior to restart for those reactors in refueling, except installation for Browns Ferry Units 1 and 2 was required by December 22, 1980. Browns Ferry had already installed continuous monitors outside the control room.

On December 2, 1980, the CMS on the SDV at Dresden Unit 2 failed to respond as expected following a normal reactor scram. Prior to this malfunction, the CMS had been made operable and tested in accordance with the Confirmatory Order described above. The BWR licensees were notified of this event by Inspection and Enforcement Information Notice No. 80-43 (Ref. B-9), which was issued on December 5, 1980. Subsequently, on December 18, 1980, the NRC staff proposed a series of actions in Inspection and Enforcement Bulletin No. 80-17 Supplement No. 4 (Ref. B-10) to improve the reliability of the CMS. Review of various CMS failures identified the lack of in situ testing to demonstrate the operability of ultrasonic systems used by the CMS to continuously monitor for water in the SDV. Supplement No. 4 provided assurance that the CMS is tested to demonstrate operability as installed, remains operable during plant operation, and is periodically surveillance tested to demonstrate ongoing operability. The Supplement stipulated that if the CMS is not operable within 30 days, the plant shall be shut down. As of early January 1981, many plants completed the required operability testing; plants that experienced scrams since the operability testing have not reported failures of the monitoring system. Licensee actions are being verified by NRC inspections.

Generic Safety Evaluation Report (SER)

The NRC staff issued the BWR Scram Discharge System Safety Evaluation report on December 1, 1980 and forwarded it for information to affected utilities on December 9, 1980 (Ref. B-11). The report specified acceptable bases for continued BWR plant operation and provided recommendations for short-term and long-term modifications. It included general design criteria to the licensees for the long-term modifications to the SDV system. These criteria were proposed by the BWR Owners' Sub-Group, and were endorsed in the SER with two additional requirements.

The first added requirement addresses the potential for fast fill of the SDV in a decaying air system pressure situation. An automatic air header dump will be required to initiate control rod insertion on low pressure in the control air header. This should prevent loss of scram function during low probability loss of air pressure events. Two months was allowed for installation of such a system.

The second added requirement deals with the SDV level instrumentation and addresses potential common-cause failures. This requirement, together with the criteria for scram system design, provides an acceptable basis for scram discharge system design (new plants) and design modifications (operating plants).

Implementation of Short-Term and Long-Term Actions

Orders to 17 operating BWRs with inadequate Scram Discharge Volume - Instrument Volume (SDV-IV) hydraulic coupling were issued on January 9, 1981, to require implementation within 90 days of provisions for automatic shutdown of the reactor when control air system pressure falls to within 10 psi of the pressure necessary for actuation of the air-operated scram outlet valves. This interim measure is to prevent sustained low pressure in the control air system for causing scram outlet valve opening in such a manner as to cause the SDV to fill and cause the loss of scram operability.

Since the Generic SER recommended this system be installed within 2 months, a supplemental SER was issued on January 12, 1981, to justify an extended time for installation, and forwarded to affected utilities on January 23, 1981 (Ref. B-12). Final disposition of the SER recommendations for long-term modifications is under consideration by the NRC.

Further reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

The following events are described below because they may possibly be perceived by the public to be of public health significance. None of the events involved a major reduction in the level of protection provided for public health or safety; therefore, they are not reportable as abnormal occurrences.

1. Show Cause Order - South Texas Project Units 1 and 2

On April 30, 1980, the NRC's Office of Inspection and Enforcement issued a Show Cause Order and Civil Penalty to Houston Lighting and Power Company, holder of the Construction Permits for South Texas Project (STP) Unit Nos. 1 and 2, when problems were identified that demonstrated inadequacies in the licensee's quality assurance program. The units, which are under construction, will utilize pressurized water reactors and are located in Matagorda County, Texas.

The facility had a history of allegations of threats and intimidation to the Quality Control inspectors from workers at the project. The NRC Region IV office received and investigated numerous allegations during 1978 and 1979 which were primarily directed toward quality assurance/quality control (QA/QC) activities. During this period, some significant civil/structural problems were identified by the licensee, and five Immediate Action Letters were issued by Region IV confirming licensee-imposed stop-work orders and corrective actions on both concrete and Cadwelding activities.

A special investigation was conducted over the period of November 10, 1979 to February 7, 1980, to determine the validity of allegations made late in 1979 and to assess the effectiveness of the QA/QC program. Most of the allegations of harassment and lack of support of the QC inspectors were substantiated and demonstrated shortcomings in the management attitude and practices at the STP. Further, the results of the investigation established that implementation of the QA/QC program at STP was deficient. These findings resulted in the issuance of a Notice of Proposed Imposition of Civil Penalties in the cumulative amount of $100,000 on April 30, 1980 (Ref. C-1). In response to the Notice of Violation, on May 23, 1980 the licensee confirmed, with minor exceptions, the items of noncompliance and forwarded payment of the $100,000 civil penalty (Ref. C-2).

On July 28, 1980, in response to the Order to Show Cause, the licensee addressed corrective actions for all items specified in the Order (Ref. C-3). The commitments for corrective actions have been carefully reviewed by the NRC and implementation in regard to the QA/QC program and limited restart of work has been under close surveillance by Region IV inspectors.

The deficiencies were found during inspections performed for a plant under construction. Most of the allegations investigated during the special investigation of November 10, 1979 to February 7, 1980, were substantiated, and although these matters did involve a number of inadequacies in management procedural and programmatic controls, no major deficiencies were found in any of the construction already completed. The procedural and programmatic inadequacies identified did not result in a major reduction in the degree of protection of the public health or safety. Therefore, the event is not considered reportable as an abnormal occurrence.

2. Inadvertent Isolation of Auxiliary Feedwater System Water Supply

On May 20, 1980, Calvert Cliffs Unit 1, a pressurized water reactor that is located in Calvert County, Maryland and is operated by the Baltimore Gas and Electric Company (licensee), was manually tripped from full power at 6:03 p.m. due to a degradation in the service water system which supplies cooling for power conversion equipment such as the main turbine, the main feedwater water pump turbines, and the safety-related emergency diesel generators and containment coolers. With the reactor in hot standby, the auxiliary feedwater pumps were being used to maintain water level in the steam generators. Due to a requirement to maintain a minimum level in condensate storage tank (CST) No. 12, the valves were realigned to take suction from CST No. 11 at approximately 8:30 p.m.

At approximately 9:30 p.m., a steam generator main feedwater pump was started to remove decay heat and the auxiliary feedwater pumps (AFP) were shut down. At approximately 10:00 p.m., the control room operator (CRO) directed that the auxiliary feedwater (AFW) valve lineup be restored to its original positions with suction taken from CST No. 12. However, when the CRO logged the operating instructions to align the AFW valves, he inadvertently transposed the nomenclature and valve numbers. This resulted in isolating both condensate storage tanks from the AFW system for Unit 1.

During routine rounds at 1:00 a.m., May 21, an operator in the turbine building discovered during taking a routine 4-hour reading that there was zero suction pressure on the AFP indicating that the suction side valve lineup was incorrect. The senior control room operator (SCRO) ordered the valve lineup to be verified and the error was corrected within 15 minutes after discovery of the condition.

In this incident, the plant had been shut down for about four hours before the AFW supply was isolated. If a loss of main feedwater had occurred, there was adequate time for operator action to restore the AFW system to service as the decay heat load had decreased. During the event there was no associated accident or radioactivity release. However, if the plant were at full power and a loss of feedwater transient (an event of moderate frequency) occurred while the AFW system was inoperable, the event would necessitate prompt corrective actions by the plant operators to restore auxiliary feedwater flow to the steam generators for decay heat removal. In such a hypothetical event, the steam generators would likely dry out in about 15 minutes ending their normal decay heat removal capability and their utility as a steam supply to the AFW system turbine-driven pumps, unless prompt operator action is taken. Feedwater transients and small-break loss-of-coolant accidents are discussed on a generic basis in NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants" (Ref. C-4). The resolution of safety concerns are still under active review.

The May 20 event identified that the licensee had insufficient controls to ensure that the valve was open when required by the plant's technical specifications. In addition, although the pressure gauge indicated the misalignment problem, it is an inadequate check in that a partially open isolation valve from the CST could give pressure readings indicating proper system alignment when, in fact, it may not exist. A similar inadvertent isolation of the AFW system water supply was found on December 3, 1975 while the plant was operating at full power. The supply valve from the #12 condensate storage tank was apparently mispositioned during the plant startup on November 19, 1975.

As corrective actions, the licensee has expanded the management and administrative controls by requiring additional verifications to ensure that changes made to valve lineups are correct. The procedures have been revised to include the use of a checklist which requires a two-man verification, one of whom must hold a senior operator license. A partial system diagram depicting the AFW suction header valves and the relationship to the AFPs and CSTs was posted in the vicinity of each valve. In addition, whenever changes are made to the auxiliary feedwater valve lineup, the suction pressure to the AFPs will continue to be checked. A similar two-man verification procedure was also instituted for controlling the repositioning of locked safety-related valves.

As the result of previous engineering evaluations, the licensee has completed short-term actions to upgrade the reliability of the service water and auxiliary feedwater systems and plans to implement additional design changes to further upgrade the system.

The NRC investigated this event as part of its inspection of plant activities for the period of May 1-31, 1980. As a result of the inspection, it was concluded that the inoperability of the auxiliary feedwater system revealed an apparent weakness in control of equipment essential to safety. It was also noted that the inoperability was neither reported to the NRC nor logged in plant records as required. Based on these and other noncompliances related to fire protection and security items, a proposed civil penalty of $21,000 was imposed on August 1, 1980 (Ref. C-5). The licensee responded on August 21, 1980; the noncompliances were not contested and the civil penalty was paid in full (Ref. C-6). The implementaion of corrective actions proposed by the licensee is being reviewed by the NRC through its licensing and inspection activities.

The May 20 event was not classified as a major reduction in the level of protection as the control room operators were trained to respond to the loss of all feedwater events, procedures existed for the operators to follow, and there was adequate time for operator corrective actions since the plant had been shut down for four hours lengthening the times for response noted above.

3. Radioactive Material in an Unrestricted Area

On September 29, 1980 the New Jersey State Department of Environmental Protection received a letter from a concerned citizen alleging he had found radioactive contamination in an area near Route 17 in Rochelle Park, New Jersey. Survey and soil sample analysis by the New Jersey State Department of Environmental Protection on October 8, 1980 and October 20, 1980, respectively, confirmed the presence of radioactive contamination in the soil samples. On November 5, 1980, NRC Region I office was informed by the New Jersey Bureau of Radiation Protection concerning their findings at this site.

On November 13, 1980, two Region I inspectors met with New Jersey State Bureau of Radiation Protection personnel at the involved site and proceeded to make radiation surveys and take soil samples. The inspectors obtained instrument readings of 1.0-3.5 millirem per hour at waist level throughout the involved area. Visual inspection of soil samples indicated the presence of compacted white and yellow material approximately one inch below the surface of the ground. Soil samples were returned to the regional laboratory for further analysis. The radioactive contamination was identified as natural thorium. Inspectors estimate that an area of approximately 200,000 square feet is involved. This area is bounded on the south by a commercial distribution firm, on the east by a chemical processing plant (Stepan Chemical Co., an NRC licensee, who has buried natural thorium on its property), on the north by railroad tracks and on the west by a field which adjoins private residential properties. The contamination appears to be confined to an area within one hundred feet of either side of Route 17 near the boundary between Maywood and Rochelle Park. Residents state that children use this area for riding trail bikes.

It appears that activities of the Maywood Chemical/Stepan Chemical Companies, which occurred prior to ACE/NRC licensing (1954), may be the source of the radioactive material, but this has not been definitely established at this time. The Maywood Chemical Company processed source material from the turn of the century until its takeover by the Stepan Chemical Company. A meeting was held with the mayors of Maywood and Rochelle Park, a representative of the New Jersey Bureau of Radiation Protection and Stepan Chemical to inform the local officials of the findings thus far and the NRC's plans for further investigation and surveys, including an overflight to determine the extent of offsite contamination. Public meetings were held to respond to public concerns. Stepan Chemical Co. has restricted access to contaminated areas on its property.

The NRC is continuing its investigation to determine the extent of offsite contamination and to ensure that necessary corrective actions are taken. There has been considerable public interest in this event. However, the levels of radioactivity involved are low and below the threshold for reporting as an abnormal occurrence.

4. Failure to Adequately Implement a Post-TMI-Action Item

During an NRC Health Physics Appraisal conducted at Nine Mile Point,* October 3-10, 1980, the licensee's commitment to NUREG-0578 (Ref. C-7), Item 2.1.8.b, "Increased Range of Radiation Monitors," was reviewed. The item, which discusses the necessity for nuclear power plants to have the capability to monitor and quantify high-level releases of noble gas in the post-accident situation, recommends the acquisition of equipment to establish an installed capability to monitor noble gases up to 10^5 μCi/cc (Xe-133). A letter to All

*A boiling water reactor plant, operated by Niagara Mohawk Power Corporation, and located in Oswego County, New York.

Operating Nuclear Power Plants from the Division of Operating Reactors, dated September 13, 1979, directed all operating reactor licensees to implement the actions contained in NUREG-0578 as soon as possible in accordance with the implementation schedule attached. A letter to All Operating Nuclear Power Plants, from the Director, Office of Nuclear Reactor Regulation, dated October 30, 1979, provided further clarification of the requirements and, in the case of Item 2.1.8.b, expanded the specification to include certain interim requirements ("provisional fix") to quantify noble gas release as high as 10,000 Ci/sec until final installation of the extended range monitors. Such requirements were to be implemented by January 1, 1980. Provided in this letter were several specific actions that were required of the licensee to satisfactorily fulfill the requirements.

The licensee in its letter dated December 31, 1979 stated that by January 1, 1980 an instrument meeting the requirements of Item 2.1.8.b would be installed within a shielded cave; that the instrument would be properly calibrated; and that a communication system would be established sufficient to allow the radiation information to be transmitted to the control room every 15 minutes.

In order to assure that all the NUREG-0578 actions required would be completed on schedule, an Order to Show Cause was issued to the licensee on January 2, 1980 that further mandated the completion of the items specified. In a sworn answer to this Order on January 22, 1980, the Executive Vice President of Niagara Mohawk Corporation affirmed that Item 2.1.8.b was completed on December 31, 1979.

On October 8, 1980, NRC Health Physics Inspectors determined that the licensee had not performed all the actions specified in the licensee's letter of December 31, 1979 for Item 2.1.8.b. Instead, the licensee had made a token effort which was technically inadequate and insufficient to meet the requirements specified. Consequently, the licensee's actions pertaining to this item were determined to be incomplete, contrary to the Executive Vice President's statement of January 22, 1980. Immediate corrective action was taken by the licensee to provide the appropriate hardware.

Subsequent investigation revealed that certain key management personnel had been aware that the licensee's actual performance in this area was substantially different from the representation provided in the licensee's December 31, 1979 letter.

As a result of evaluation and analysis of this occurrence, on November 26, 1980, the NRC Office of Inspection and Enforcement issued a proposed civil penalty of $225,000 based on the items of noncompliance identified and issued an Order for Modification of License (Effective Immediately) and Order to Show Cause (Ref. C-8). The Order required the licensee to (1) separate the (then) site superintendent from further involvement with nuclear matters for Niagara Mohawk Power Corporation; (2) implement procedures to ensure that managers at all levels of the licensee organization provide full, accurate and timely information to higher management and to the NRC when submitted thereto; and (3) show cause why the Executive Vice President should also not be removed from involvement in nuclear matters.

The NRC Inspection and Investigation Reports dated December 2, 1980 presented evidence that indicated that the licensee's management control system was not effective in establishing and maintaining reasonable assurance that information provided to higher management and to the Commission was accurate and reliable. Lack of formalized procedures detailing the process by which management controls were maintained and implemented contributed significantly to the inability of the licensee to control and monitor this particular activity.

In response to the NRC enforcement letter issued November 26, 1980, the licensee submitted replies on December 19 and 29, 1980 and January 3 and 22, 1981 which described actions being taken and which took exception to certain of the NRC findings. Based on these submittals, several meetings between representatives of the licensee and NRC personnel, and a further submittal by the licensee on March 20, 1981, the NRC issued to the licensee a Withdrawal of Ordered Modifications and Order to Show Cause and Termination of Proceedings Thereon on March 20, 1981 (Ref. C-9). The action reduced the penalty to $215,000, which the licensee has agreed to pay, and withdrew the prohibitions against the (then) site superintendent and the proposed prohibitions against the Executive Vice President. It was determined that the mistaken statement made by the (then) site superintendent resulted from poor management control over the flow of information within the licensee's organization and not from an intent to willfully withhold or deceive. In addition, it was determined that the licensee had provided adequate cause for allowing the Executive Vice President to continue involvement with nuclear matters, in that he relied on the management chain of the licensee in signing a letter to NRC; the corporate staff relied on the information submitted from the plant site which it did not question based on the concurrence of the site staff; that there was no intent by the Executive Vice President to deceive the NRC; and that the company has adopted procedures that should prevent recurrence of further erroneous submittals.

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B-1. U.S. Nuclear Regulatory Commission, "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," USNRC Report NUREG-0313, Revision 1, July 1980.*

B-2. U.S. Nuclear Regulatory Commission, "Annual Report." These annual reports summarize the major activities of the NRC and are submitted to the President for transmittal to Congress.**

B-3. U.S. Nuclear Regulatory Commission, "Unresolved Safety Issues Summary, Aqua Book," USNRC Report NUREG-0606, issued quarterly as a series.*

B-4. U.S. Nuclear Regulatory Commission, "Mark I Containment Short-Term Program Safety Evaluation Report," USNRC Report NUREG-0408, December 1977.*

B-5 U.S. Nuclear Regulatory Commission, "Mark I Containment Long-Term Program Safety Evaluation Reoprt," USNRC Report NUREG-0661, July 1980.*

B-6 U.S. Nuclear Regulatory Commission, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," USNRC Report NUREG-0619, November 1980.*

B-7 The fire in the electrical cable trays at Browns Ferry was originally reported as an abnormal occurrence in USNRC Report NUREG-75/090, October 1975 ("Report to Congress on Abnormal Occurrences, January-June 1975").* The incident was closed out in USNRC Report NUREG-0090-4, October 1976 ("Report to Congress on Abnormal Occurrences, April-June 1976").*

B-8 U.S. Nuclear Regulatory Commission, "Draft Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station Unit 2," USNRC Draft Report NUREG-0683, July 1980.***

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*Available for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555 and National Technical Information Service, Springfield, Virginia 22161.

**Available for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

***Single copies available from USNRC Division of Technical Information and Document Control, Washington, DC 20555.

REFERENCES (continued)

B-9 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 80-43, "Failure of the Continuous Water Level Monitor for the Scram Discharge Volume at Dresden Unit No. 2," December 5, 1980.†

B-10 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR," December 18, 1980.†

B-11 Letters from D. Eisenhut, NRC, to BWR licensees, forwarding "Generic Safety Evaluation Report, BWR Scram Discharge System (dated December 1, 1980)," December 9, 1980.†

B-12 Letters from D. Eisenhut, NRC, to BWR licensees, forwarding "Generic Safety Evaluation Report, BWR Scram Discharge Systems, Supplement 1 (dated January 12, 1981)," January 23, 1981.†

C-1 Letter from Victor Stello, Jr., NRC, to G. W. Oprea, Jr., Houston Lighting and Power Company, forwarding a Notice of Violation, Notice of Proposed Imposition of Civil Penalties, and an Order to Show Cause, Docket Nos. 50-498 and 50-499, April 30, 1980.†

C-2 Letter from G. W. Oprea, Jr., Houston Lighting and Power Company, to Victor Stello, Jr., NRC, Docket Nos. 50-498 and 50-499, May 23, 1980.†

C-3 Letter from G. W. Oprea, Jr., Houston Lighting and Power Company, to Victor Stello, Jr., NRC, Docket Nos. 50-498 and 50-499, July 28, 1980.†

C-4 U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," USNRC Report NUREG-0635, January 1980.**

C-5 Letter from Victor Stello, Jr., NRC, to George V. McGowan, Baltimore Gas and Electric Company, "Combined Inspections 50-317/80-06 and 50-318/80-06," Docket Nos. 50-317 and 50-318, August 1, 1980. (Appendix C of this letter is a Notice of Proposed Imposition of Civil Penalties.)†

C-6 Letter from George V. McGowan, Baltimore Gas and Electric Company, to Victor Stello, Jr., NRC, Docket Nos. 50-317 and 50-318, August 21, 1980.†

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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 for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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C-7 U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," USNRC Report NUREG-0578, July 1979.**

C-8 Letter from Victor Stello, Jr., NRC, to William J. Donlon, Niagara Mohawk Power Corporation, forwarding a Notice of Violation, Notice of Proposed Imposition of Civil Penalties, and an Order for Modification of License (Effective Immediately) and Order to Show Cause, Docket No. 50-220, November 26, 1980.†

C-9 Letter from Victor Stello, Jr., NRC, to William J. Donlon, Niagara Mohawk Power Corporation, forwarding a Withdrawal of Ordered Modification and Order to Show Cause and Termination of Proceedings Thereon, Docket No. 50-220, March 20, 1981.†

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

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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 or safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1980.

There was one abnormal occurrence at the nuclear power plants licensed to operate; the event involved significant flooding of a reactor containment building. There were no abnormal occurrences at the fuel cycle facilities (other than nuclear power plants). There were no abnormal occurrences at other licensee facilities. There were no abnormal occurrences reported by the Agreement States.

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

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