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

OFFICE OF THE
CHAIRMAN

May 5, 1980

COMMISSION
CORRESPONDENCE

The Honorable Thomas P. O'Neill, Jr.
Speaker of the United States
House of Representatives
Washington, D.C. 20515

Dear Mr. Speaker:

We submit herewith the nineteenth 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 fourth calendar quarter of 1979.

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. These incidents or events, including any submitted by the Agreement States, are as follows:

1. There were no abnormal occurrences at the nuclear power plants licensed to operate.

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

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

4. There was one abnormal occurrence reported by the Agreement States. The incident involved overexposure of a hot cell operator.

This report also contains information updating some previously reported abnormal occurrences, including an update on the nuclear accident at Three Mile Island.

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,

John F. Ahearne
John F. Ahearne
Chairman

Enclosure:
Report to Congress
on Abnormal Occurrences

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NUREG-0090
Vol. 2, No. 4

Report to Congress on Abnormal Occurrences

October - December 1979

Office of
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Report to Congress on Abnormal Occurrences

October - December 1979

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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, the nineteenth in the series, covers the period from October 1 to December 31, 1979.

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

1. There were no abnormal occurrences at the nuclear power plants licensed to operate.

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

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

4. There was one abnormal occurrence reported by the Agreement States. The incident involved overexposure of a hot cell operator.

This report also contains information updating some previously reported abnormal occurrences, including an update on the nuclear accident at Three Mile Island.

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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. 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 action (e.g., violations, infractions, deficiencies, 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, the nineteenth in the series, covers the period between October 1 - December 31, 1979.

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.

Most NRC licensee employees who work with 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 in the NRC and the industry in their continuing evaluation and improvement of nuclear safety. 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 any 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 1979

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth quarter of 1979. As of the end of the reporting period, the NRC had not determined that any events were abnormal occurrences.

FUEL CYCLE FACILITIES

(Other Than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth quarter of 1979. As of the end of the reporting period, 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 fourth quarter of 1979. As of the end of the reporting period, the NRC had not determined that any events were abnormal occurrences.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth quarter of 1979, an Agreement State reported the following abnormal occurrence to the NRC.

AS79-5 Overexposure of a Hot Cell Operator

Appendix A (Example 1 "For All Licensees") of this report notes that exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On August 30, 1979, at the Gamma Industries facility in Baton Rouge, Louisiana.

Nature and Probable Consequences - A hot cell operator received sufficient dose to produce blistering of several fingers and the thumbs of both hands. He had noticed tenderness and slight thickening of the skin on the thumb and index finger of his right hand on September 7, 1979. The conditions worsened, and other fingers became involved before the individual was examined at Ochsner Foundation Hospital in New Orleans on September 12, 1979. At that time he was diagnosed as having an allergic reaction to nickel. However, because of his work with radioactive materials, the individual was examined at the Radiation Emergency Assistance Center/Training Site (REAC/TS) facility in Oak Ridge, Tennessee on September 21, 1979, where technetium-99m pertechnetate scans of the hands were performed and cytogenetic tests conducted. From this examination it was estimated that he received a dose of from 2,500 to 3,000 rads to the thumb, index, middle, and ring finger of the right hand and to the thumb and first two fingers of the left hand. The individual's TLD (thermo-luminescent dosimeter) badge indicated a whole body dose of 600 mrem. At the time of this report, the individual has returned to work and has apparently recovered from the acute effects of the overexposure.

Cause or Causes - On August 30, 1979, an 8,000 Curie shipment of iridium-192 pellets had been unloaded. The bulk shipping container held several hundred unencapsulated iridium-192 pellets. After unloading the bulk container, an attempt was made to replace the container's top using the remote manipulators. After ensuring that there were no pellets left in the container or on the counter in the hot cell, the individual entered the hot cell and replaced the top by hand. Radiation surveys were required and performed using 3 monitoring systems, but did not detect radiation emanating from iridium-192 fragments at the base of the partially capped shipping container. It is the opinion of the Louisiana Nuclear Energy Divisions' Staff that improper handling of the shipping container was the most probable cause of the excessive dose.

Actions Taken to Prevent Recurrence

Louisiana Nuclear Energy Division - A violation for the excessive exposure was cited, and a review of the hot cell procedures was performed by the Division. From this review it was established that the hot cell operator had received training in the proper procedures and was aware that he should not handle the shipping capsule except with the manipulators. A recommendation was made that an operating manual for hot cell operators be provided for use in the training phase of hot cell operators to document the training provided.

Licensee - The Division has received written notification of the following immediate steps which have been taken as a result of this accident:

1. All hot cell operators have been instructed that no person shall attempt to replace the caps on shipping capsules directly. All shipping capsules will be made ready for disposal by use of hot cell manipulators.

2. All hot cell operators will be monitored by wrist TLD's in an attempt to evaluate extremity doses and evaluate operations that might lead to excessive extremity doses.

This incident is closed for the purpose of this report.

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 (42 FR 10950) on February 24, 1977.

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.

<u>For Commercial Nuclear Power Plants</u>

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 1979 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, and Vol. 2, No. 2. It is further updated as follows:

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

The NRC staff updated the implementation document NUREG-0313 as a subtask under Generic Task A-42, "Pipe Cracks in Boiling Water Reactors." The objective of other subtasks is to identify and recommend additional measures to reduce the susceptibility of stainless steel piping to stress corrosion cracking. A report (NUREG-0313, Rev. 1) on the results of this task was published in October 1979. Following the issuance of this report, the NRC noticed the availability of this report in the Federal Register and requested interested parties to provide any comments to the NRC by January 16, 1980. To date, comments have been received from six organizations. The staff is evaluating these comments and a further revision to NUREG-0313, Rev. 1 is expected to be issued in June 1980.

The measures outlined in Part II of NUREG-0313, Rev. 1 provide for positive actions that are consistent with current technology. The implementation of these actions should markedly reduce the susceptibility of stainless steel piping to stress corrosion cracking in BWRs. It is recognized that additional means could be used to limit the extent of corrosion of BWR pressure boundary piping materials and to improve the overall system integrity. These include plant design and operational procedure considerations to reduce system exposure to potentially aggressive environment, improved material selection, special fabrication and welding techniques, and provisions for volumetric inspection capability in the design of weld joints. The use of such means to limit intergranular stress corrosion cracking (IGSCC) will be reviewed on a case-by-case basis.

Although the items identified below are not required for the present plant safety, they may be expected to leau to means of limiting the extent of IGSCC and improving the change of detecting such IGSCC. Some of these items have not yet been fully developed (or have recently been developed) and have not yet been accepted by the NRC. Specifically, areas that require further consideration are:

a) Improved ultrasonic inspection methods. Such methods should be included in the ASME Code or included in a Regulatory Guide.

b) Development and implementation of an improved focused inservice inspection program based on stress rule index, material of construction, history of cracking, etc.

c) Improved weld joint design to ensure that required examinations can be performed effectively.

d) Reduction of oxygen content in reactor coolant during all phases of reactor operation by water chemistry control, de-aeration of systems, etc.

e) Minimization of stagnant or low flow coolant pressure boundary piping.

f) Evaluation of newly developed alternate corrosion resistant materials in BWR environment.

g) Evaluation of improvement of material corrosion resistance by alternate methods such as heat sink welding, induction heating stress improvement, etc.

h) Evaluation of the Electrochemical potentiokinetic reactivation technique for detecting and quantifying the degree of sensitization in stainless steel piping.

i) Continued evaluation and verification of leak before break concept.

j) Evaluation and implementation of leakage detection capability to improve early detection of small leaks.

Further reports will be made as appropriate.

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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, and Vol. 2, No. 3. It is further updated as follows:

76-11 Steam Generator Tube Integrity

Since the last general update of this item (NUREG-0090, Vol. 1, No. 4), the following significant developments related to pressurized water reactor (PWR) steam generator tube integrity have occurred.

Westinghouse Designed Units

Degradation of steam generator tubes, due to a corrosion-related phenomenon known as "denting," has continued at Surry Unit 1, Turkey Point Units 3 and 4, and Indian Point 3. Denting at Connecticut Yankee, R. E. Ginna, Indian Point Unit 2, Point Beach Units 1 and 2, H. B. Robinson, and San Onofre Unit 1 has somewhat stabilized. Minor denting at North Anna Unit 1 was discovered during a November 1979 inspection.

Point Beach Unit 1 and R. E. Ginna continued to experience small steam generator tube leaks due to a phenomenon designated as "deep crevice cracking."

The steam generator replacement program at Surry Unit 2 is essentially complete. Replacement of the Surry Unit 1 steam generators was scheduled to begin on June 1, 1980; however, this date may slip due to potential delays caused by preparation of an Environmental Impact Statement. Replacement programs fo. Turkey Point Units 3 and 4 have been reviewed and a Safety Evaluation Report published; hearings on these replacement programs are anticipated. A Westinghouse topical report on their in-situ retubing concept is under review.

A recent update of this item (NUREG-0090, Vol. 2, No. 3) reported on a steam generator tube rupture which occurred at Prairie Island Unit 1 due to a spring, inadvertently left inside the steam generator following sludge lancing, rubbing against the tube during operation. No problems with steam generator tube integrity have been encountered since the unit returned to power.

Trojan experienced a steam generator tube leak due to a defect tangent to the inner row U-bend. Similar defect indications were found at North Anna Unit 1 in two tubes and, though not yet confirmed, Farley Unit 1.

Combustion Engineering (CE) Designed Units

Degradation of steam generator tubes due to "denting" has continued at St. Lucie Unit 1. Plans to chemically clean the steam generators, previously considered for April 1979, were not implemented and have not been rescheduled. Further development is necessary before chemical cleaning procedures are implemented (e.g., development of an effective solvent). Modifications to the steam generators, discussed in a previous report on this item, have stabilized denting at Maine Yankee and Millstone Unit 2. However, support plate cracking has apparently accelerated at Millstone Unit 2.

A proposed steam generator replacement program for Palisades is under review.

Babcock & Wilcox (B&W) Designed Units

Oconee Unit 1 has not experienced a steam generator tube leak due to flow-induced vibration of tubes located adjacent to the open inspection lane in over a year and a half. During the December 1979 inspection, degraded tubes were also identified away from the inspection lane mainly at the 14th tube support plate. A total of 68 tubes required plugging. Only one tube leak has occurred in the past year in a B&W unit. The leak occurred in an off lane tube in the B steam generator at Oconee Unit 1 in July 1979.

Details of the experience by the three PWR reactor designers since the last general update report (NUREG-0090, Vol. 1, No. 4) are described below.

Westinghouse

The cause of steam generator tube "denting" at North Anna Unit 1 is unique compared to the other affected units in this country. An inadvertent dumping of 200-300 pounds of resin from the polishers of the full flow demineralizers into the North Anna Unit 1 steam generators caused sulfuric acid to be produced which attacked the carbon steel support plates. A cleanup procedure developed by Westinghouse and VEPCO is under review. For the reasons set forth in previous reports, minor "denting" is not considered a significant safety hazard.

The "deep crevice cracking" problem previously reported has accelerated rapidly at Point Beach Unit 1 over the past 12 months with small leaks occurring on March 1, August 5, August 29, and December 11, 1979. Because of the constraint provided by the tubesheet, deep crevice cracks are not considered a significant safety concern. However, due to the rapidly spreading extent and magnitude of the problem at this unit, additional operating conditions were imposed on the license to assure continued safe operation. The operation of this unit is being carefully monitored by the NRC.

The defects in the tangent to the U-bend experienced in tubes at Trojan, North Anna Unit 1, and possibly Farley Unit 1 are located in row 1 tubes. These defects are believed to be cracks resulting from corrosion since the row 1 tubes have the highest residual stresses due to fabrication. As a result of the cracking, North Anna Unit 1 has plugged all row 1 tubes and installed an inspection port for future observation. Trojan has committed to pull the affected tubes during the forthcoming late spring 1980 refueling outage and subject them to laboratory tests.

Combustion Engineering

The "denting" at Maine Yankee and Millstone Unit 2 has been stabilized by the removal of lugs and portions of the solid rim in the uppermost support plates. Support plate cracking has apparently accelerated at Millstone Unit 2. These modifications have not been made in the St. Lucie Unit 1 steam generators, due to the small amount of tube denting; however, the level of denting continues to increase slightly. Florida Power & Light Company had proposed a chemical cleaning process intended to remove the corrosion products from the tube/tube support plate crevices, but the process has yet to be applied.

Babcock & Wilcox

B&W has not positively identified the initiating mechanism for the circumferential cracking of the tubes along the open tube lane. One hypothesis is that moisture carry-over dries out at the 14th-15th tube support plate leaving a chemical deposit and accompanying small metal ions. Chemical attack, under the deposit, acts as a crack initiator and along with the tube vibrations results in crack propagation. Many design and operating modifications have been implemented by Duke Power at the Oconee units. No fatigue cracks have occurred in over a year and a half. The degradation identified in the off-lane tubes is believed to be caused by erosion with some chemical attack and particle impingement and related to local thermohydraulic conditions. The chemical attack and particle impingement is a result of "debris" located on the support plates. The origin of the iron oxide is not known.

NRC Actions

The NRC staff continues to closely monitor, review and evaluate, and approve the acceptability of continued operation of plants experiencing steam generator tube problems. A number of generic reviews and studies have been undertaken as part of three generic tasks in the NRC Program for the Resolution of Generic Issues. Specifically, the generic Task Action Plans A-3, A-4, and A-5 are directed at the particular problems of Westinghouse, Combustion Engineering, and Babcock and Wilcox. These Task Action Plans are scheduled to be completed in May 1980.

Under these tasks, generic studies will be conducted to (1) evaluate inservice inspection results from operating reactors, (2) evaluate the consequences of tube failures under postulated accident conditions, (3) evaluate tube structural integrity, (4) establish tube plugging criteria based on new information, (5) define the requirements for monitoring secondary coolant chemistry, (6) evaluate inservice inspection methods, and (7) review design improvements proposed for new plants.

Further reports will be made as appropriate.

* * * * * *

The following abnormal occurrence was originally reported in NUREG-0090-10, "Report to Congress on Abnormal Occurrences: October-December 1977," and updated in a subsequent report in this series, i.e., Vol. 1, No. 4. It is further updated as follows:

77-8 Generic Design Deficiency

On December 1, 1979 Beaver Valley Unit 1 was shut down for refueling and extensive modifications. Part of these modifications are associated with the permanent design changes to correct the net positive suction head (NPSH) problem and include the following:

1. Cold quench spray (QS) water from each QS header will be diverted to the sump or intake area of the containment recirculation spray (CRS) pumps to assure an adequate amount of NPSH for the CRS pumps both inside and outside containment. The cold QS water will lower the vapor pressure of the water entering the pump. Orifices in the new lines will limit the flow.

2. The recirculation spray and QS nozzles will be replaced with more efficient nozzles to improve the performance of the sprays.

3. A loop seal in the piping will be added to each of the existing QS flow paths to prevent the refueling water storage tank (RWST) from draining to the sump if, after a loss-of-coolant accident (LOCA), a QS pump is not in operation and the valves along the flowpath are open.

4. Two trains from the chemical addition tank (CAT) will be added, each with two pumps to deliver known quantities of sodium hydroxide to the QS system. This will assure a correct sump pH to control radioactive iodine in containment.

Surry Units 1 and 2 modifications proposed are:

1. Diverting a portion of the cold quench spray (QS) water from each of the QS headers to each of the outside recirculation spray (RS) pump suction piping.

2. Routing a bleed flow from the discharge of the RS cooler back to the suction of the respective inside RS pump.

3. Installing flow restricting orifices in each line to ensure correct flows.

The NRC staff has almost completed its review. Beaver Valley Unit 1 will have the modifications in place before restart in July 1980. Surry Unit 2 modifications are planned to be in place prior to restart in mid-year 1980 and Surry Unit 1 during the steam generator repair outage to start in summer 1980.

Further reports will be made as appropriate.

* * * * * *

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

78-5 Loss of Containment Integrity

Review of this generic concern is continuing. In late October 1979, the NRC staff developed an Interim Position for Operability of Containment Vent and

Purge Valves. This position was sent to licensees of all operating reactors with a request for response within 45 days. Licensee responses are currently under NRC staff revew. From these responses, the NRC staff will determine whether or not the license of each reactor plant should be modified, suspended, or revoked.

Further reports will be made as appropriate.

* * * * * *

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 a subsequent report in this series, i.e., Vol. 2, No. 2. It is further updated as follows:

79-2 Deficiencies in Piping Design

As previously reported, the Nuclear Regulatory Commission ordered five plants to shut down on March 13, 1979, until reanalysis and necessary modifications were made to safety-related piping systems to bring them into conformance with requirements for withstanding earthquakes. The plants ordered to shut down were Beaver Valley Unit 1, James A. FitzPatrick, Maine Yankee and Surry Units 1 and 2. The problem pertained to use of an algebraic summation method to combine seismic forces in a computer code which resulted in prediction of stresses significantly lower than would be predicted by NRC approved techniques.

The required reanalysis and necessary modifications were completed for Maine Yankee and Beaver Valley and orders were issued on May 24, 1979, and August 8, 1979, respectively, terminating the March 13, 1979 Show Cause Orders. Sufficient reanalysis and modifications were completed for FitzPatrick and Surry Unit 1 to permit issuing orders on August 14, 1979, and August 22, 1979, respectively, allowing resumption of operation for 60 days while some remaining pipe support analyses were completed. Based on licensees' submittals, the 60-day requirements were satisfied by FitzPatrick and Surry Unit 1 in that the remaining pipe support analyses had been completed and the schedules for implementing the necessary modifications had been made.

Surry Unit 2 was shut down for steam generator repair and replacement prior to the March 13, 1979 shutdown order. Because of the long shutdown for steam generator work, the seismic reanalysis required by the order was delayed by the licensee. It is not anticipated that the required seismic reanalysis will lengthen the plant shutdown.

NRC IE Bulletin 79-07 was issued April 14, 1979 requiring all licensees to review the computer codes used in the design of safety-related systems to determine if the algebraic summation technique had been used. All affected operating units have been resolved completely, or based upon NRC staff evaluation, have been permitted to continue operation during reanalysis. In each case where continued operation was permitted, the reanalysis methods utilized and the margin in the original piping design to code allowable values were such that few modifications to piping systems were necessary due to algebraic summation.

Also, as previously reported, an additional issue was identified which involved the accuracy of the information input for seismic analyses. NRC IE Bulletin 79-14 was issued on July 2, 1979 (revised on July 19, 1979 and supplemented on August 15 and September 7, 1979) to all power reactor facilities wit· an operating license or a construction permit. The Bulletin directed the licensees to perform inspections of their safety-related piping systems and supports. Because of the conservatism and redundancy built into the piping systems, the NRC did not require the facilities to be shut down pending completion of the inspections and remedial action if required. However, several plants, including Ft. St. Vrain, Millstone Unit 2, D. C. Cook Unit 1, and Rancho Seco, have shut down for various lengths of time pursuant to technical specification requirements as a result of discrepancies discovered during the inspections. The majority of all operating plants have had to modify and/or add supports due to deviations identified between existing "as-built" field conditions and design documents. Although the inspections and NRC review of the results are not yet complete, it appears all but one or two facilities will require some hardware fixes as a result of inspections performed pursuant to this IE Bulletin. Due to the amount of effort involved, refueling schedules, and lack of qualified stress analysts and pipe support engineers, the time frame for completion of this Bulletin has slipped, with the majority of plants scheduled to satisfy Bulletin requirements by April 1980.

Further reports will be made as appropriate.

* * * * *

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 the series, i.e., NUREG-0090, Vol. 2, No. 2 and Vol. 2, No. 3. It is further updated as follows:

79-3 Nuclear Accident at Three Mile Island

Remote Access into the TMI-2 Reactor Containment Building

Two remote accesses into the TMI-2 reactor containment building were made in August and November 1979. The first remote access was achieved by boring a 4 inch diameter opening into the inner cover plate of a spare containment penetration. A total of 6 waste water samples from the TMI-2 reactor containment building were obtained through this penetration, and samples were analyzed for radioactivity concentrations.

The radiochemical analyses indicated the predominant isotopes of cesium-137 (175 μCi/cc), cesium-139 (40 μCi/cc), strontium-90 (45 μCi/cc), and tritium (1 μCi/cc).

The second remote access was achieved by boring a 9 inch diameter opening into the inner cover plate of a spare containment penetration. Through this penetration, the TMI-2 containment building remote surveillance was accomplished by (1) inserting a TV camera for a visual scan of the area, (2) inserting radiation surveillance monitors to evaluate beta and gamma fields, (3) obtaining air samples of the containment atmosphere, (4) obtaining samples of horizontal and vertical surface contamination using swiping techniques, and (5) inserting radiation survey equipment enshrouded with samples of radiation protective clothing for measurements of shielding effectiveness.

Further reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

The following events are described below because they may possibly be perc ved by the public to be of public health significance. Neither event involved a major reduction in the level of protection provided for public health or safety; therefore, they are not reportable as abnormal occurrences.

1.0 Temporary Closing of Commercial Burial Facilities for Low-Level Waste

In the summer of 1979, attention was called to a number of instances in which certain packages of low-level waste were not in compliance with federal requirements on arrival at one of three commercial burial facilities in the country. Such items of noncompliance included a fire on a truck carrying combustible waste, leaking packages, truck contamination from improperly closed packages, free liquid in packages of supposedly dry solid material, inadequately labeled packages, and improperly documented shipments. While none of these items of noncompliance by itself represented a significant health concern, collectively they did show a lack of proper attention to federal requirements for packaging and shipping of radioactive waste materials.

The Governors of the Agreement States of South Carolina, Washington, and Nevada, in which the commercial burial facilities are located, notified the NRC of the repeated disregard for these rules and at various times closed or limited these facilities to certain shippers. The NRC, in conjunction with the U.S. Department of Transportation (DOT) determined that the Federal government should improve its assurance that federal regulations governing these shipments are met and took several steps:

- The NRC changed its regulations to specifically subject its licensees to DOT regulations and thus effectively increase the federal inspection capability.
- The NRC issued bulletins to all licensees (1) to inform licensees of the transportation incidents that occurred, of requirements for transportation of low-level radioactive waste materials, and of burial site requirements and (2) to require licensees to submit written management-approved procedures for safe transfer, packaging, and transportation of these materials.
- The NRC increased its inspections at shipper and receiver sites.
- The NRC modified its enforcement criteria to increase penalties.
- The NRC and DOT are jointly investigating ways to improve safety of low specific activity material packages.

- The NRC is acquiring support from the Society of Nuclear Medicine to improve medical waste packages and from the Atomic Industrial Forum to improve industrial waste packages.

- The NRC and DOT are making an effort to better inform shippers of requirements.

- The NRC is developing a draft regulation for burial of low-level wastes.

2.0 Turbine Disc Cracking

On November 5, 1979, Wisconsin Electric Power Company, in a meeting on another subject, notified the NRC of cracking in the keyway areas of Westinghouse manufactured low pressure steam turbines. Subsequently, on November 17, 1979, an anonymous letter was received, informing the Commission of an October 30, 1979, meeting between Westinghouse and utility owners having Westinghouse turbines. The purpose of the meeting was to inform the utility owners of the cracking problem.

On November 20, 1979, the Westinghouse Steam Turbine Division confirmed the existence of bore cracking, in addition to keyway cracking, during an inspection of the Zion Unit 1 low pressure turbine. Prior to this date, keyway cracking had been observed in Point Beach Unit 1 by means of a newly developed ultrasonic inspection technique. Impetus for the development of the technique was the discovery of keyway cracking in Surry Unit 2 turbine discs by magnetic particle inspection.

The primary NRC concern, since the turbines are not safety related, has been the possibility of the generation of missiles which might cause a breach of the containment. This is a postulated concern, as the only known disc failure in a U.S. nuclear turbine did not penetrate the turbine housing and thus generated no external missiles. The NRC is currently evaluating the potential for other problems resulting from a turbine failure.

On December 17, 1979, the NRC was presented with an updated version of the information which had been provided to the utility executives at the October 30, 1979 meeting. This was followed by a proprietary submittal dated December 20, 1979, providing much of the technical background data which led to the request for early shutdown for inspection of Indian Point Unit 3.

The NRC is currently requiring turbine inspections on a timely basis to prevent disc failures, and is continuing to collect data and evaluate the extent of the problem. In addition to Westinghouse turbines, the turbines of other manufacturers are being included in consideration of this problem.

On December 28, 1979, the NRC issued IE Information Notice No. 79-37 informing licensees of the cracking problems.

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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, the nineteenth in the series, covers the period October 1 to December 31, 1979.

During the period, there was one abnormal occurrence. The event occurred at an Agreement State licensee and involved overexposure of a hot cell operator.

This report also contains information updating previously reported abnormal occurrences.

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